Second United Nations
International Conference
on the Peaceful Uses of
Atomic Energy

A/CONF.15/P/1782
U.S.A.
June 1958

ORIGINAL: ENGLISH

CONSTRUCTION DESIGN OF EBR-II: AN INTEGRATED UNMODERATED NUCLEAR POWER PLANT

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I. INTRODUCTION

The Argonne Experimental Breeder Reactor II (EBR-II) is an unmoderated, heterogeneous, sodium-cooled reactor and power plant with a power output of 62.5 megawatts (mw) of heat. The energy produced in the reactor is converted to 20 mw of electricity through a conventional steam cycle. The reactor is fueled with U^{235} or plutonium, and the plant includes an integral fuel processing facility where the irradiated fuel is processed, fabricated, and assembled for return to the reactor.

The preliminary design of this plant was described at the First International Conference on the Peaceful Uses of Atomic Energy in 1955 (Paper No. P/501). Although the basic design concept and objectives have remained unchanged, many modifications have been made in the detailed design, including separation of the facility into several plants and providing containment for the reactor system.

The separations process employed permits the buildup of certain fission products; operation of the plant will determine the effect of buildup of these fission products, as well as the buildup of the higher isotopes of uranium and plutonium.

The EBR-II is primarily an engineering facility to determine the feasibility of this type of reactor for central station power plant application. Major emphasis has been placed on achieving high thermal performance at high temperatures, and high fuel burnup with a fast and economical fuel cycle. The thermal performance of the reactor and the size of the system components are such as to permit direct extrapolation to central station

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application. The plant has been designed to permit a maximum of experimental operational flexibility by separation of the plant systems, and yet permit extrapolation to a commercial plant which would not require the same degree of separation.

The EBR-II Facility consists of several buildings, each of them housing a major part of the plant. The general arrangement of the Facility is shown in Figure 1. The reactor and primary sodium coolant system are housed in the Reactor Plant which is a steel containment vessel 80 ft in diameter and approximately 140 ft high. The Power Plant contains the turbine generator and associated steam and electrical equipment. It also contains the control room for the reactor and power cycle. The Process Plant contains the facilities for processing and fabrication of the fuel. The Sodium Plant houses the secondary sodium pump and associated equipment, and the Boiler Plant houses the sodium to water steam generator. These buildings are interconnected with the Reactor Plant and Power Plant by sodium or steam piping as shown.

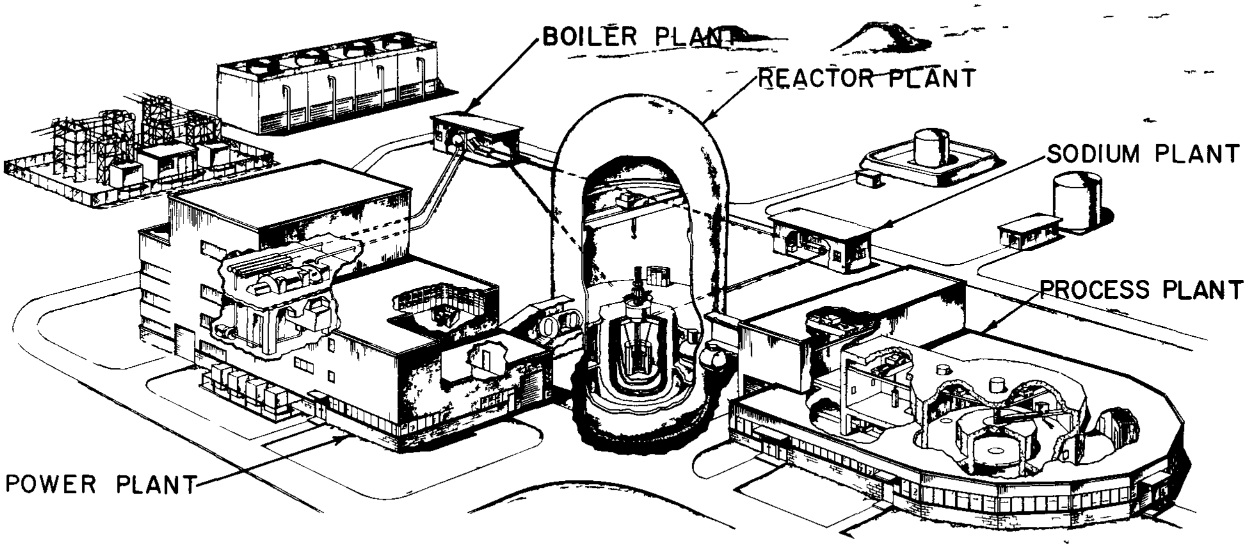


Fig. 1. EBR-II Plant

Construction of the plant was started late last year at the National Reactor Testing Station in Idaho. The construction design of the plant has been proceeding concurrently and at the time of this meeting is nearing completion. Construction of the reactor building containment shell is scheduled for completion in October. Other construction is in progress.

Major construction is scheduled for completion by April 1960. Preoperational testing and critical experiments will be initiated at that time. Operation of the reactor and power system will follow.

The present estimate of cost of the plant is $29,000,000. This includes engineering and construction of the entire facility. The plant is being constructed on a new site and includes all general site development and supporting facilities as well as the reactor, complete power cycle, and fuel cycle facilities.

A. The Power Cycle

Heat is removed from the reactor by the primary sodium coolant system and transferred to the secondary sodium system in a shell-and-tube heat exchanger. The secondary system transfers the heat to the steam generator where superheated steam is produced to drive a conventional turbine-generator.

The primary sodium flow rate is 8200 gpm, entering the reactor at 700°F and leaving at 900°F. The secondary sodium flow rate is 6050 gpm, entering the heat exchanger at 610°F and leaving at 880°F. The secondary sodium enters the superheater at 880°F, leaves at 808°F, enters the evaporator at this temperature, and leaves at 610°F. 249,000 lb/hr of feedwater is supplied to the evaporator at a temperature of 550°F. Saturated steam at 1310 psig (580°F) flows through the separator and superheater, leaving at 850°F. The superheated steam is employed to drive the turbine-generator, to drive the feed-water pump turbine, for direct feed-water heating, and to maintain a minimum by-pass flow around the turbine-generator to the condenser. A full capacity steam by-pass system is provided to permit reactor operation without turbine-generator operation, or with turbine load at any fraction of reactor power. Feed-water heating is accomplished by extraction from the main turbine, the exhaust from the feed-water pump turbine, and high pressure steam from the main steam line.

The three heat transfer systems are shown in Fig. 2 which is a skeleton flow sheet of the power cycle. It is based on a reactor power output of 62.5 mw, and 5000 lb/hr of steam by-pass to the condenser.

B. The Fuel Cycle

Irradiated fuel from the reactor will be cooled for only 15 days prior to transfer to the Process Plant for decanning and processing. The processing facilities are contained in two shielded cells located in the Process Plant. The larger of these cells contains the equipment for decanning spent fuel and blanket elements, for processing them, and for fabricating the fuel into new elements. This cell is gastight and contains an inert atmosphere of high-purity argon. The second cell is a conventional shielded hot cell with an air atmosphere and is used for assembly of subassembly units as well as service work for the main cell.

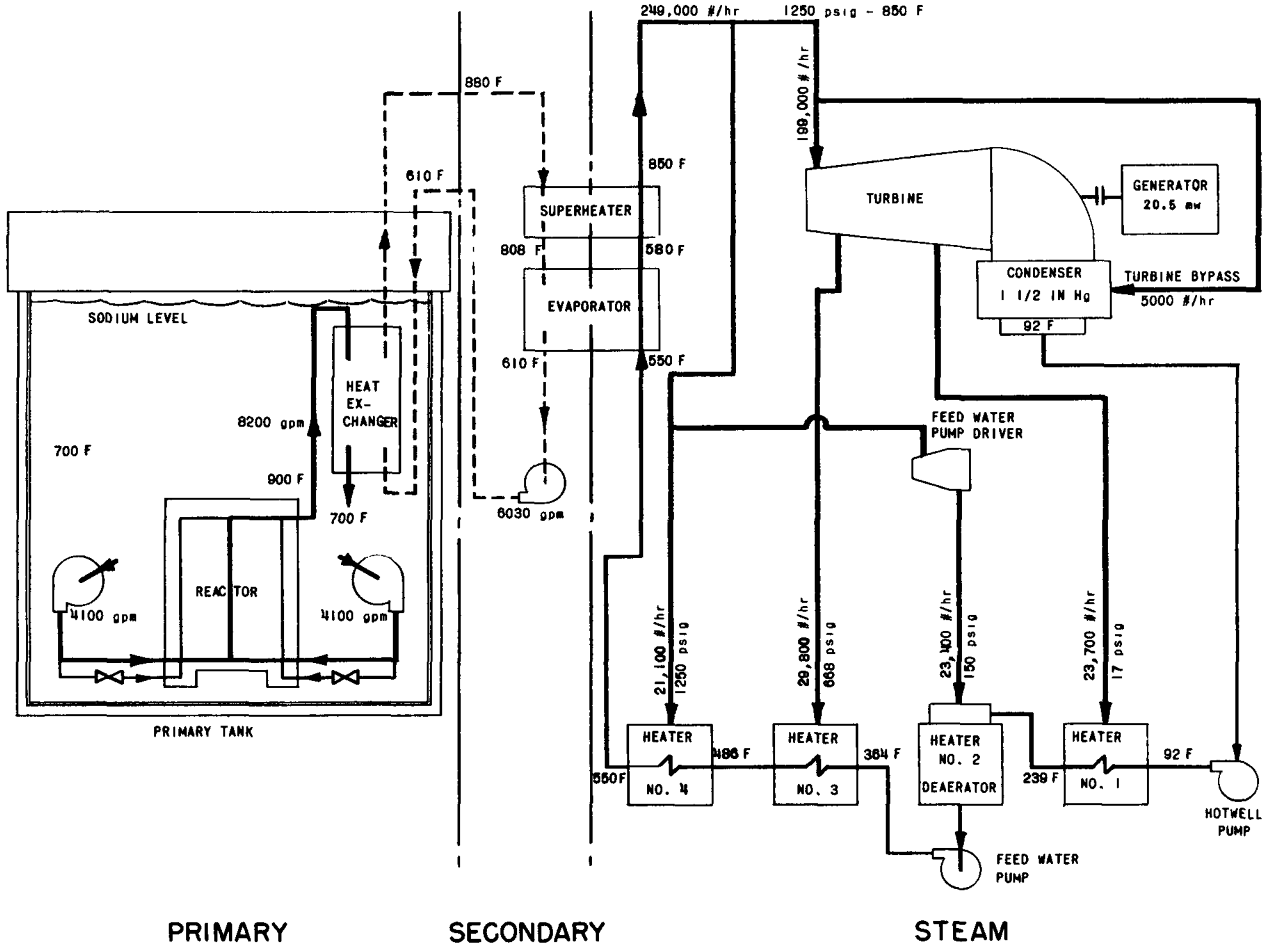


Fig. 2. EBR-II Skeleton Flow Diagram

Irradiated fuel assemblies are disassembled in the air cell and the individual fuel elements are transferred to the inert gas cell. Here the elements are mechanically decanned. The bare fuel pins are then charged to a furnace in 10-kilogram batches and melted in an environment deficient in oxygen. Under these conditions, the volatile and noble gas fission products are released to the furnace atmosphere and those fission products whose oxides are more stable than uranium oxide (cerium, rare earths) appear in the dross or slag. The ingot resulting from this melting operation contains uranium, plutonium, and the fission products Zr, Nb, Mo, Ru, Rh, Pd and presumably Tc. These elements reach an equilibrium value, and it is the "estimated equilibrium" alloy that will be used for the EBR-II fuel. The ingot produced by the processing furnace is remelted in an injection-casting furnace and new fuel pins are cast directly to size in expendable Vycor molds. The pins are then cut to length and inspected prior to reassembly into new elements.

The fuel pins are then assembled into new fuel elements, sodium bonded, and tested. The finished elements are transferred from the argon atmosphere cell to the air cell where they are assembled into new subassemblies for return to the reactor.

The details of this process cycle and the design of the Process Plant and facilities are described in detail in another paper prepared for this conference.(1)

II. PRIMARY SYSTEM

The primary system is located in the Reactor Plant and includes the following:

Reactor
Primary Cooling System
Control and Safety Drive Systems
Fuel Handling System

They are contained within a single, double-walled vessel or "Primary Tank" 26 ft in diameter and 26 ft in depth. The reactor, the major components in the primary cooling system, and parts of the drive systems and fuel handling system are located below the liquid level of the bulk sodium contained therein and operate submerged. A blanket of argon gas is maintained above the free surface of the sodium in the tank.

A. Reactor

The reactor consists of the various subassemblies of fuel and blanket elements, the grid-plenum chamber assembly, the reactor vessel, and the reactor cover as shown in Fig. 3. The reactor is divided into three main radial zones: core, inner blanket, and outer blanket. Twelve control rods are located at the outer edge of the core, and two safety rods are located within the core as shown in Fig. 4.

Subassemblies

Each radial zone of the reactor is comprised of right hexagonal subassemblies 2.29 in. across external flats of the hexagon with a 0.040 in. hexagonal tube wall thickness. All subassemblies are of identical size. The subassemblies are spaced on a triangular pitch of 2.32 in. center distance. A nominal clearance of 0.030 in. between each subassembly permits removal of the units from the reactor. Each face of the core and inner blanket subassembly hexagonal tubes contains a projection or "button" 3/8 in. diameter by 0.014 in. high. The buttons are located approximately at the horizontal

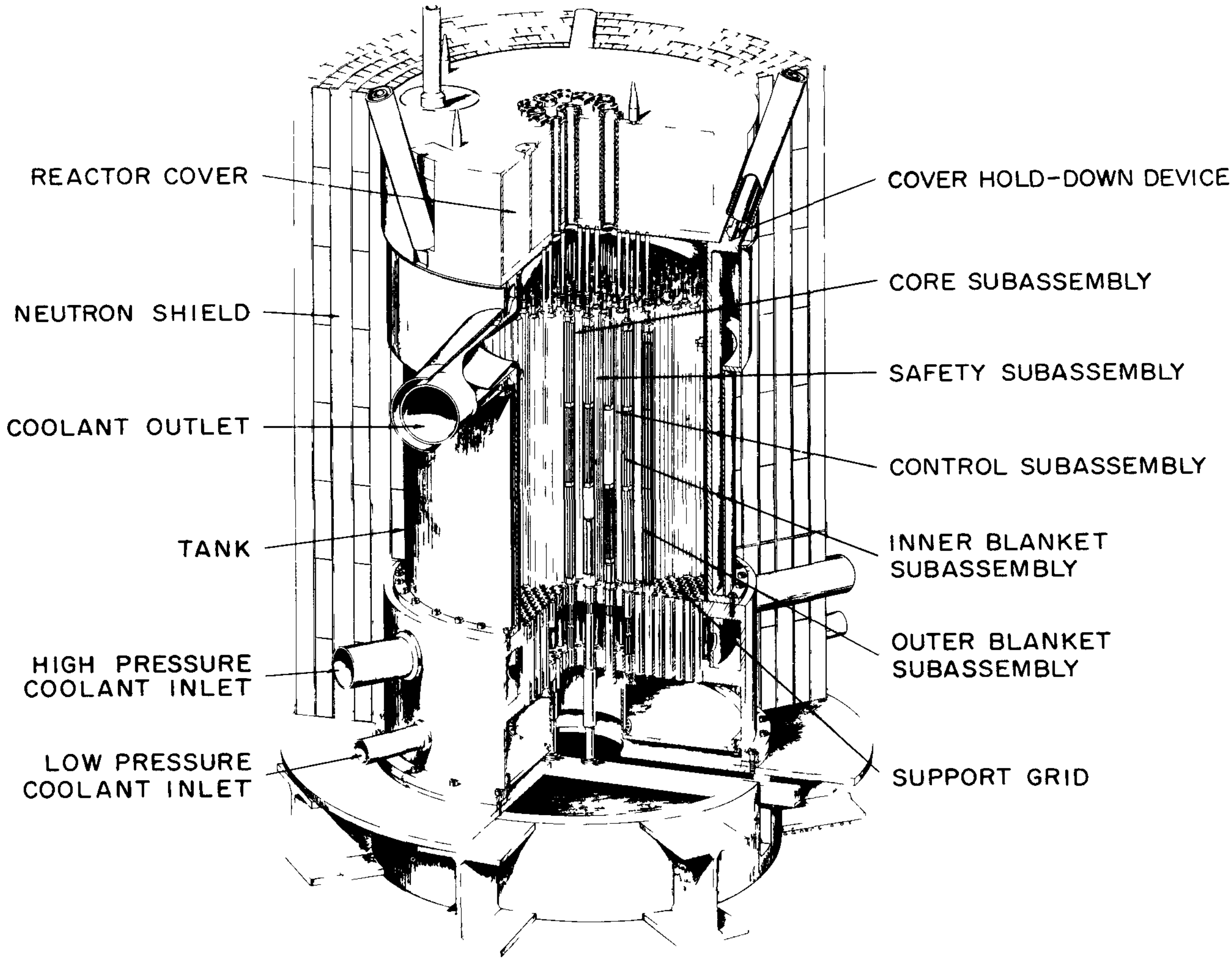


Fig. 3. EBR-II Reactor

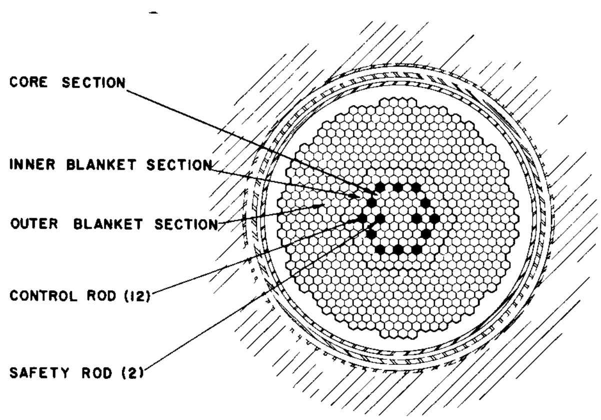


Fig. 4. Reactor Arrangement

center-line of the reactor and provide a plane of contact at that location. The upper end of each subassembly (including control and safety subassemblies) is identical. All subassemblies are accommodated by the same handling and transfer devices. Each subassembly contains a number of fuel and/or blanket elements, of size and shape appropriate to the particular type of subassembly.

The core, including the control and safety rods, has an equivalent radius of 9.52 in. (27.17 cm) and a height of 14.22 in. (36.12 cm); a total core volume of 66.3 liters. Located in the core zone are 47 core subassemblies, 2 safety subassemblies, and 12 control subassemblies. The twelve control rods and the two safety rods consist of modified movable core subassemblies. The rods, plus their stationary thimbles, comprise the control and safety subassemblies. The external dimensions and lattice spacing of the thimbles are identical to the core and blanket subassemblies.

The core subassembly (Fig. 5) is comprised of three "active" sections:

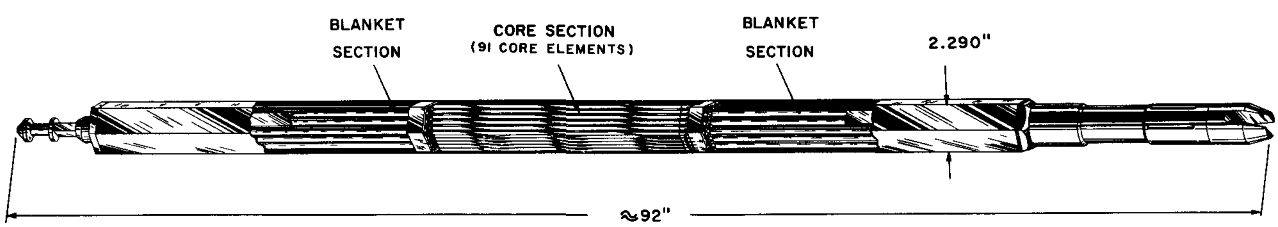


Fig. 5. EBR-II Core Subassembly

upper blanket, core, and lower blanket. The core section consists of 91 cylindrical fuel elements spaced on a triangular lattice by a single, helical rib on the outside of each element. The elements are supported within the subassembly by fastening their lower ends to a parallel strip support grid. The upper ends are unrestrained. The fuel elements are "pin type," consisting of a right circular cylinder or pin of fuel alloy (0.144 in. diameter by 14.22 in. long) fitted into a thin-walled, stainless steel tube as shown in Fig. 6. The coolant flows along the outside of the fuel element tube.

The fuel pin is contained in a stainless steel tube with a 0.006 in. sodium filled annulus between the pin and the inside of the tube to provide a thermal bond. An inert gas space is provided above the sodium to accommodate expansion of the sodium. The fuel element tube is welded closed at each end.

The upper and lower blanket sections are identical in construction and each consists of 19 pin-type elements spaced on a triangular lattice. The unalloyed depleted uranium pins are 0.316 in. diameter and total 18 in. long.

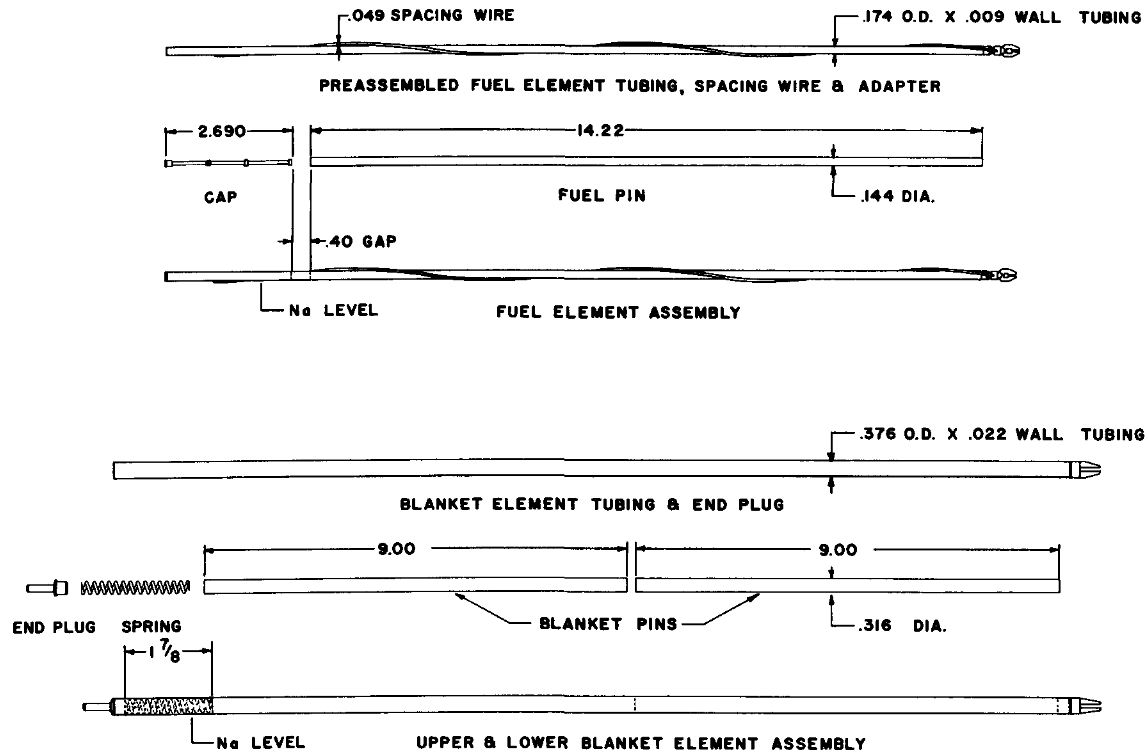


Fig. 6. Core Subassembly Elements

They are contained within a stainless steel tube with a 0.008 in. sodium filled annulus to provide the necessary thermal bond. The details of the upper and lower blanket elements are shown in Fig. 6.

Both ends of the blanket elements are positioned in the subassembly by a parallel strip grid similar to that employed for the fuel elements. Axial expansion is permitted, but other movements are restricted.

The upper adapter of the assembly is provided with an attachment knob for the various gripper units, and a collar for the transfer arm. These are provided for fuel handling purposes.

The lower adapter is a cylindrical nozzle, and serves the combined function of location and support in the reactor grid, and inlet nozzle for the coolant. The bottom end of the nozzle is closed, and coolant enters the nozzle through holes in the cylindrical wall.

The inner and outer blanket subassemblies are each comprised of 19 cylindrical blanket elements spaced on a close packed triangular pitch and contained in the hexagonal subassembly as shown in Fig. 7. The "active" blanket section consists of depleted uranium cylinders (0.433 in. diameter) totaling 55 in. in length. They are contained in a stainless steel tube with a

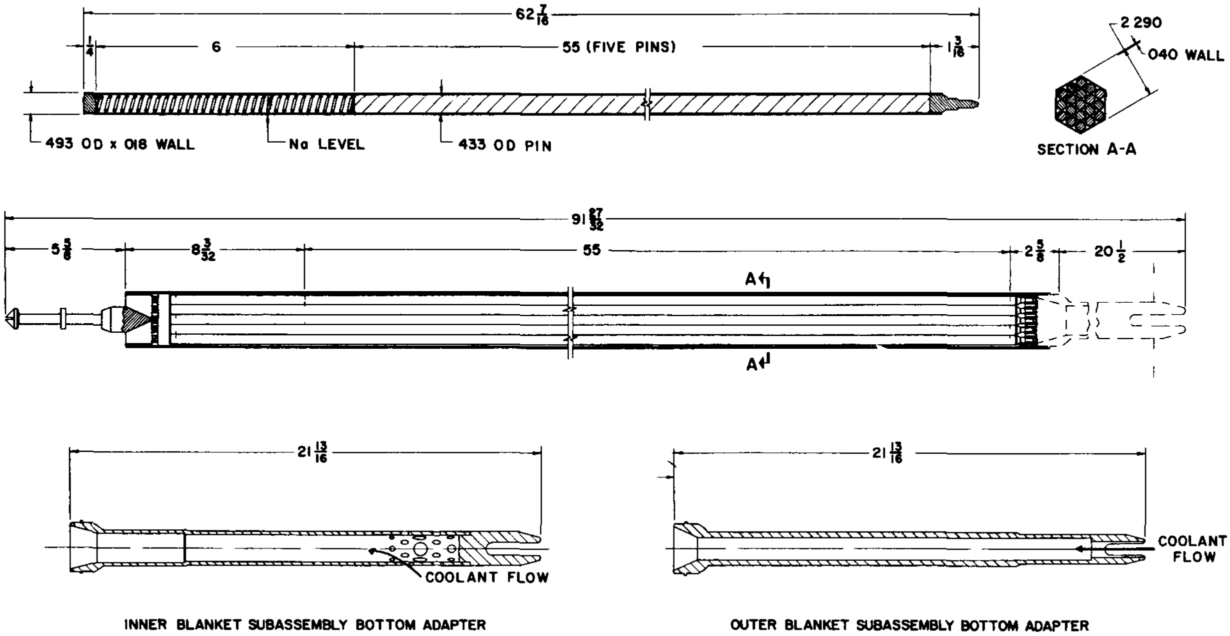


Fig. 7. EBR-II Inner Blanket and Outer Blanket Subassemblies

0.012 in. annulus filled with static sodium to provide a thermal bond and an argon gas expansion region above the sodium. The end closures are welded to provide a sealed unit.

The two types of blanket subassemblies differ only in the design of the lower adapter. The inner blanket subassembly lower adapter is similar to the lower adapter of the core subassembly, except that it is smaller in diameter. The outer blanket subassembly lower adapter is smallest in diameter and also of different design. It contains an opening at the bottom end, through which the coolant enters. The two different lower adapters employed in the subassemblies are shown in Fig. 7.

Twelve identical control rods are employed to provide operational control of the reactor. The control rod consists of a modified core subassembly with a core section comprised of 61 fuel elements identical to those employed in the core subassembly. The control rod is encased in a hexagonal tube 1.908 in. across flats, which is smaller than the hexagonal thimble tube by the equivalent of one row of fuel elements. The control rod does not contain an axial blanket. A void section equivalent in height to the reactor core is provided above the core section. During operation, this section if filled with the sodium coolant. A reflector section of solid steel (except for flow passages for the coolant), is located immediately above the void section. The lower end of the control rod below the fuel section consists of a cylindrical tube also containing a steel reflector section. Bearings are provided on this lower section which provide the guide between the control rod and the guide thimble. The control rod guide thimble is removable from the reactor in the event of damage. It is locked in the lower reactor grid by a latch which is engaged by rotating the thimble. Rotation of the thimble is normally prevented by the six subassemblies which surround it.

Two identical safety rods are employed to provide "shutdown reactivity" during reactor loading operations. The safety rod is essentially identical to the control rod except for modifications at the lower end, to provide the necessary attachment to the drives.

The guide thimble is locked to the lower reactor grid structure in a similar manner to that described for the control rod guide thimble. The safety rod is engaged to the driving mechanism by a rotational locking mechanism. Inadvertent disengagement of the safety rod is prevented by a hexagonal-shaped collar on the upper end of the safety rod. This normally engages the inside of the thimble, preventing rotation of the safety rod. To connect or disconnect the safety rod for loading purposes, the safety rod must be raised 1 in. above its normal "up position" by the safety rod drive mechanism.

Grid-Plenum Assembly

The grid-plenum assembly incorporates a grid structure which supports and locates the subassemblies and a plenum chamber arrangement which directs the inlet coolant flow. The grid consists essentially of two stainless steel plates spaced and interconnected by a large number of tubes welded to each plate within the outer blanket zone. The plates are perforated with axially aligned locating holes for the lower adapters of the subassemblies. The adapters pass through the upper plate and extend into the lower plate. The subassemblies are supported by a spherical shoulder on the subassembly which engages a conical seat at the top of the upper grid plate to provide a seal as shown in Fig. 8. This arrangement positions the subasemblies accurately, supports them securely, and minimizes leakage flow from the inlet plenum chambers to the spaces between subassemblies.

Two coolant inlet plenum chambers are provided. The high pressure coolant plenum chamber, which is the source of supply for the core and inner blanket subassemblies, is comprised of the cylindrical space between the two grid plates. The coolant enters this plenum at the periphery, flows radially inward to the core and inner blanket zones, and then enters the subassemblies through the holes in the walls of the bottom adapters near the lower grid plate. The lower end of the subassembly adapter is closed, forming a "hydraulic piston." The sodium in the high pressure plenum is at a pressure of approximately 60 psig, and this pressure acts across the piston. This provides a downward force of about 160 lb on the core subassemblies and 130 lb on the inner blanket subassemblies. This force plus the weight of the subassemblies exceeds the lifting force exerted on the subassemblies by the coolant.

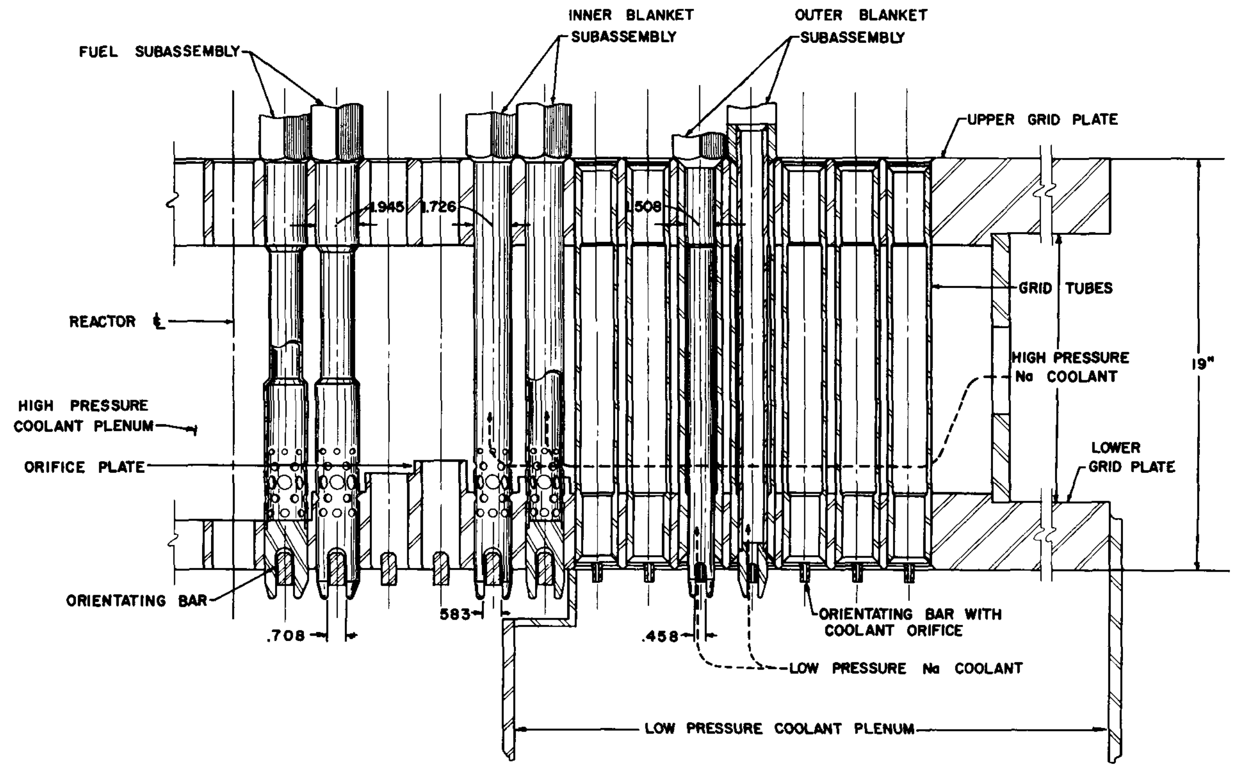


Fig. 8. Subassembly Support Grid

The upper surface of the lower grid plate is "stepped" in such a manner as to vary the number of subassembly adapter entrance holes available for introducing flow into the subassemblies. The latter arrangement provides orificing of the flow through the subassemblies to match the heat generation rate.

The control and safety rods are also cooled by the high pressure sodium coolant which enters through slots in the thimble adapters, and then through a second set of slots in the lower end of the rods. The slots in the thimble section are above the lower bearing of the rod throughout its travel. The lower end of the thimble is open, and the lower rod bearing serves as a flow restriction to prevent sodium leakage from the bottom of the thimble. The primary system sodium pressure acts across the lower end of the rod, supplying a "hold-down" force similar to the arrangement in the core subassemblies.

The low pressure coolant plenum, which supplies the outer blanket, consists of an annular chamber immediately below the lower grid plate. The coolant enters the plenum at about 17 psig, and flows into the bottom adapters of the outer blanket subassemblies through the openings in their bottom ends. Because the pressure drop through the outer blanket subassemblies is much smaller than through those of the other zones, it is unnecessary to provide "hydraulic hold-down" for them.

Three different hole diameters are employed in the grid plates. The core section has the largest diameter hole, the inner blanket section has a smaller diameter hole, and the outer blanket section has the smallest diameter hole. This arrangement prevents a fuel subassembly from being placed inadvertently in an inner blanket position or an outer blanket position and, likewise, an inner blanket subassembly cannot be placed in an outer blanket position. To prevent the interchange of subassemblies in the other direction, subassembly orientation bars are used. They are fastened to the underside of the lower grid plates as shown in Fig. 8, and engage slots in the ends of the subassembly lower adapters.

There are three thicknesses of bars: the core subassemblies engage the thickest, the inner blanket subassemblies the next thickest, and the outer blanket subassemblies the thinnest. If an inner blanket subassembly is inadvertently placed in a core position, the slot in the inner blanket subassembly tip is too narrow to allow the slot to engage the bar. This prevents engagement of the subassembly at least 2 in. short of its normal position in the grid, which condition is easily detected by the loading mechanism. A similar condition exists if an outer blanket subassembly is placed in an inner blanket position or a core position.

This arrangement of loading control is employed because a core subassembly inserted in either blanket zone introduces both a reactivity problem and a cooling problem, while a blanket subassembly introduced in the wrong zone introduces only a cooling problem. The grid structure has a total depth of 19 in. The reactor core, however, is only 14 in. long. If a subassembly adapter cannot enter the grid structure because the subassembly is in the wrong location, such loading error will not permit the fuel section of the subassembly to enter the core region. In the reverse situation, a subassembly can engage the grid for approximately 17 in. of travel, but the error is readily detectable.

Reactor Vessel

The reactor vessel is a cylindrical tank equipped with a mounting flange at either end and a single (outlet) coolant nozzle at its upper end. The vessel is positioned between the grid-plenum assembly and the removable vessel cover. Thermal barriers are provided to assure acceptable thermal stresses in the vessel wall under the most severe transient temperature conditions contemplated.

Reactor Cover

The reactor vessel cover provides the closure of the upper end of the reactor vessel and forms the upper surface of the outlet coolant plenum chamber. It also contains the upper portion of the neutron shield. The twelve control rod drive shafts operate through the cover, and guide bearings are provided in the cover for these units. During the unloading operations, the fuel gripper mechanism also operates through an opening in the cover. A small amount of leakage occurs through these various openings during reactor operation when a sodium pressure differential of approximately 10 psi exists across the cover. This leakage flow is employed as a part of the neutron shield cooling system in this region.

The top cover is raised and lowered by two shafts penetrating the small rotating plug. It is fastened to the reactor vessel by three clamping mechanisms, and the raising and lowering mechanism is so designed as to permit free expansion of the lifting shafts. This arrangement avoids the large load due to internal pressure being transferred to the cover lifting mechanism, and also avoids problems associated with differential thermal expansion in the system.

The underside of the top cover is provided with pin "projections" on the same spacing as the subassemblies. These pins are positioned directly above each subassembly adapter and provide approximately 3/16 in. of clearance between the adapter and the end of the pin. The pins do not make contact with the subassemblies, but prevent any appreciable lifting of the subassemblies in the event of malfunction of the hydraulic hold-down system.

B. Primary Cooling System

The primary system component arrangement is shown schematically in Fig. 2. The reactor vessel is centrally located at the bottom of the primary tank. The pump, heat exchanger, and connecting piping are disposed radially around the reactor vessel and elevated somewhat above it.

Coolant is supplied from the bulk sodium in the primary tank to the high pressure and low pressure inlet plenum chambers by two identical pumps operating in parallel. From these plenum chambers, the coolant passes upward through the various subassemblies, into the common outlet plenum chamber above the reactor, exits via the single outlet nozzle, and passes to and through the shell side of the shell and tube heat exchanger from whence it returns to the bulk sodium in the primary tank.

The two main sodium pumps are vertically mounted, single stage, centrifugal pumps which supply 4250 gpm of coolant (each) at 60 psi head. They utilize a hydraulic liquid sodium bearing located at the pump impeller. The direct coupled drives are totally enclosed, leakproof, 2300 volt, A-C motors. Shaft seals of a labyrinth type are provided to minimize sodium vapor in the motor enclosure. The motors are frequency controlled over about a 10 to 1 speed range, providing smooth and continuous control over the entire speed range. Flow control valves are not employed in the primary coolant system.

A 5000 gpm, 40 psi head, centrifugal type pump of a design similar to that intended for use in the primary system has been operated for more than 7500 hrs at speeds of 1750, 890, and 175 rpm and at temperatures up to 900°F (Fig. 9).

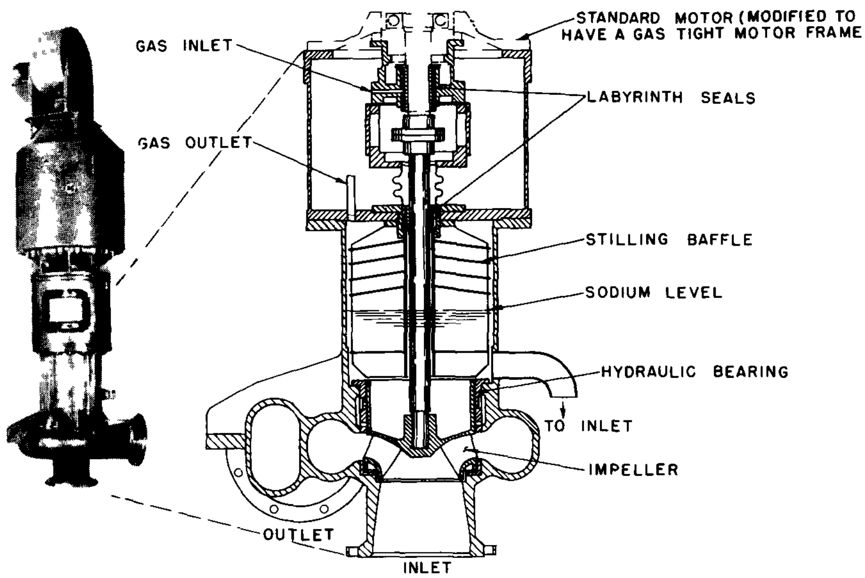
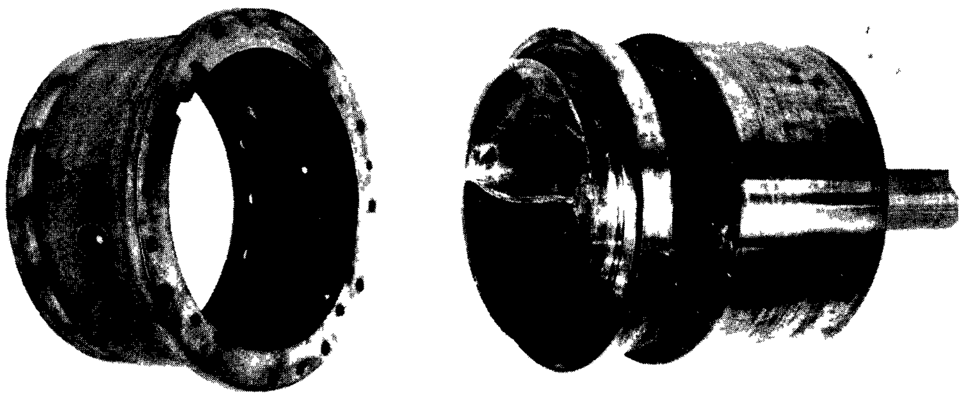


Fig. 9. 5000 gpm Mechanical Sodium Pump

This pump has been subjected to 126 starts, 26 of which occurred during low speed operation. After 6500 hrs of operation, the pump was disassembled and thoroughly inspected. The pump was in excellent condition, and the hydraulic bearing showed no indication of wear (Fig. 10). The pump, its performance, and operating experience are described in detail in another paper prepared for this conference.(2)



Bearing Shell

Bearing Journal and Impeller

Fig. 10. Hydraulic Bearing and Journal after 6500 hrs of Operation

In order that the pumps can be removed from the primary tank for maintenance, ball-seat type pipe disconnects are used in the lines between the pumps and the piping to the reactor inlet plenum chambers. The heat exchanger tube bundle and associated structure are removable as a unit in a vertical direction; however, the heat exchanger shell is permanently attached to the cover of the primary tank (Fig. 11). The sodium line between the upper plenum of the reactor and the heat exchanger shell is permanently attached between these two components.

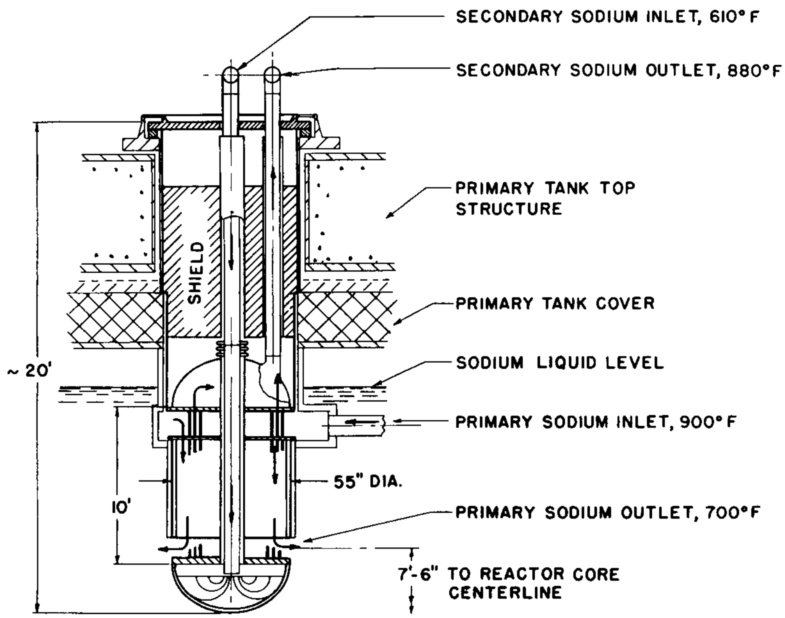


Fig. 11. EBR-II Sodium to Sodium Heat Exchanger.

In this sodium line is located the "Auxiliary Pump." The auxiliary pump operating in series with the two main sodium pumps is a permanent magnet D-C electromagnetic pump having a capacity of 500 gpm at 0.2 psi head and a 900°F sodium temperature. At this capacity, the pump requires 8100 amperes at 1.0 volt. The pumping section is incorporated in the 14 in., schedule 20, reactor outlet pipe with no change in pipe cross-section (Fig. 12). The heavy electrical leads make no physical contact with the pipe, but rather make electrical contact through a sodium filled container. This relatively light weight, double-walled, electrically insulated container is welded directly to the pipe wall. This design allows an electrical short circuit path between the two bus bars of 800 amperes, which is insignificant considering the extreme reliability of this design and the small power requirements.

The auxiliary pump electrical power is supplied from metallic rectifier units and storage batteries. The storage batteries, operating in parallel with the rectifier units, assure pump operation in the event of a complete power failure. During normal operation, these batteries float on the line and

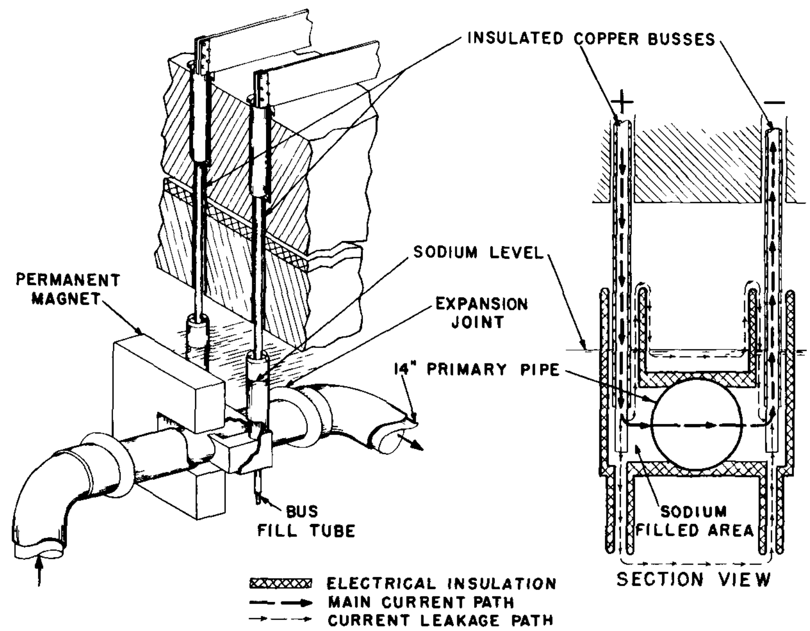


Fig. 12. EBR-II Primary System Auxiliary Pump

remain fully charged at all times. In the event of a sustained power failure, the pump operates until the batteries are discharged, which results in a gradual decay of the flow rate and an ideal "transition" to thermal convection.

Iron-nickel-alkaline cells (14 in parallel) are employed because of their ability to sustain repeated discharging. The discharge characteristics of the batteries and corresponding pump flow characteristics are shown in Fig. 13.

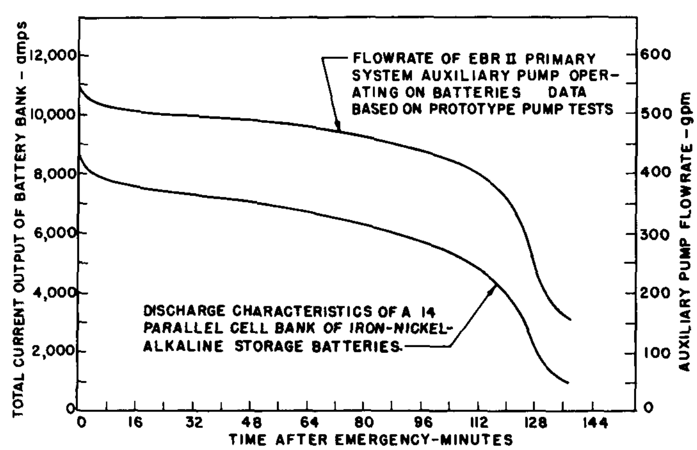


Fig. 13. Battery Current and Pump Flow Characteristics of Auxiliary Pump System

Since the no-load voltage of these batteries (also the trickle charge voltage) is 1.57 volts, the electrical system is designed for this voltage. The actual pump voltage across the pump bus is only 80 millivolts; the remainder of the voltage drop is taken in the power supply system. The pump is rated at 1.0 volt because this is the battery voltage shortly after pumping begins on battery power. When operating from the rectifier power supply at 1.57 volts, the pump current is 12,800 amperes. This is reduced approximately 10%, due to back emf, when the main pumps are in operation.

Interlocks between the auxiliary pump and reactor controls prevent reactor startup unless the pump is connected and operating with the batteries fully charged.

The primary purpose of the auxiliary pump is to augment thermal convection under certain conditions of reactor shutdown. These conditions occur as a result of system malfunctions which tend to "destroy" the temperature distributions necessary to maintain thermal convection. The auxiliary pump insures continuity of flow under these conditions and prevents undesirable temperature transients.

A prototype model of the D-C electromagnetic auxiliary pump has operated for more than 2700 hrs (Fig. 14). In this test the sodium was at a temperature of 900°F, and the magnet assembly was at 750°F. These conditions duplicate those to be encountered in the EBR-II. The Alnico 5 magnet supplied a flux of 1000 gauss in the 16 in. gap (which decreased approximately 5% from room temperature to 750°F). There was no change in the magnetic characteristics of the permanent magnet throughout the test. The pump performed extremely well and as anticipated.

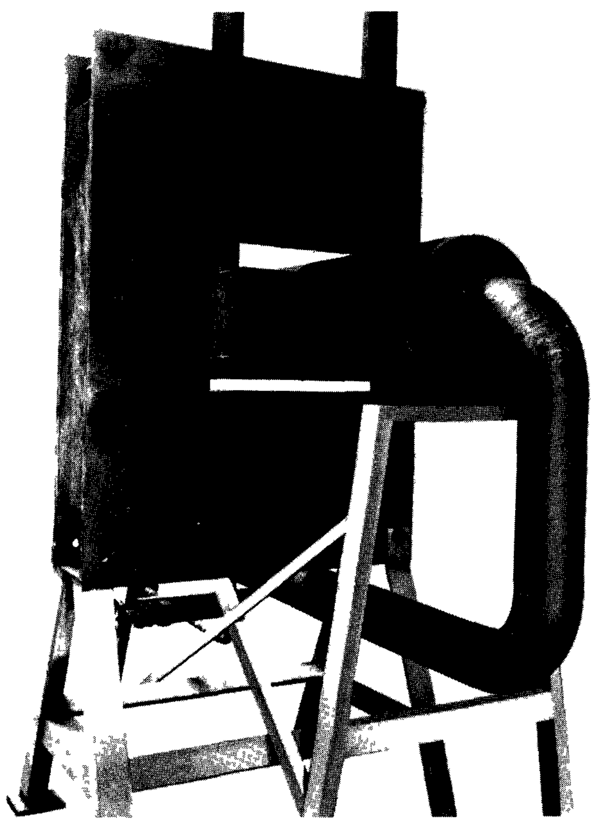


Fig. 14. Auxiliary Pump and Test Loop

C. Control and Safety Drive Mechanism

The EBR-II is controlled by moving fuel in the reactor. Operation of the reactor is controlled by twelve control rods. Each rod is independently driven by an electrical-mechanical drive mechanism (Fig. 15). The drives are identical and are so arranged that only one drive may be operated at a time (with the exception of "scram," when all twelve operate simulataneously). Operating control is achieved by a 14 in. vertical motion of the control rods which is provided by a rack and pinion type drive with constant speed electric motors; therefore, only one speed of movement is possible.

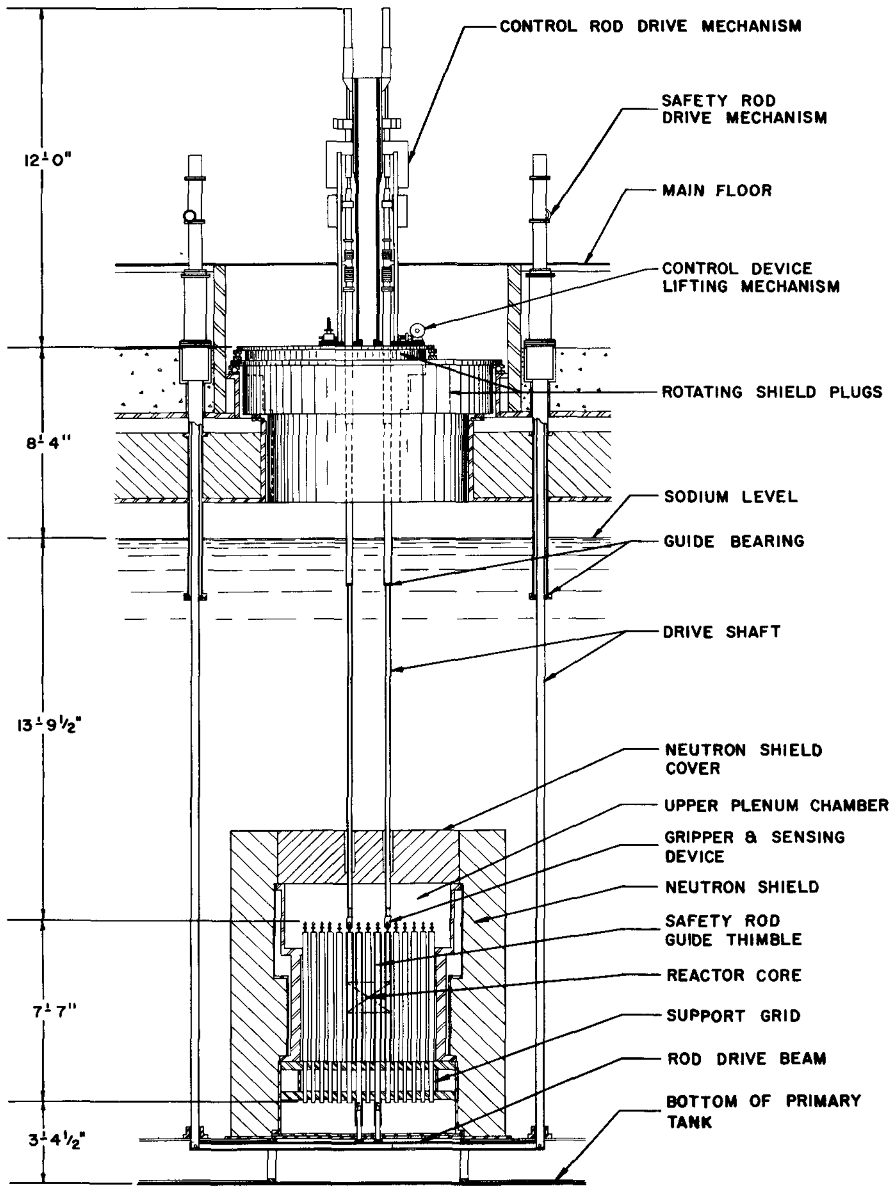


Fig. 15. EBR-II Control and Safety Rod Drive System

The control rods are disconnected from their drives during fuel loading operations. The disconnect is made with the control rods in their down or least reactive position. The control rods remain in this position during the unloading procedure.

Although the twelve control rods and drives are identical, during steady-state reactor operation eleven are used for shimming and one is used for regulating. The "regulating rod" is defined, for descriptive purposes, as the rod being controlled by the automatic control system; any of the twelve rods may be used as the regulating rod. During manual operation, either during change in reactor power level or steady-state operation, all twelve rods are defined as "shim rods." The power supply is arranged to supply power to only one shim rod drive unit at a time, restricting rod movement to one rod at a time. At steady-state reactor operation, with the regulating rod on automatic control, one shim rod can be moved to permit adjusting the position of the regulating rod in the reactor.

Drive speed (either up or down) is mechanically limited to 5 in./min. Since the total worth of each rod will not be more than 0.006 $\Delta k/k$, the drive speed available restricts the maximum possible reactivity addition rate to less than 0.00011 $\Delta k/k$ per sec (based on two rods).

During scram, the twelve control rods are ejected (downward) from the core by air pressure plus gravity. Rod release time, or time between receipt of scram signal at the rod drive and start of rod movement is 0.008 sec.

Two safety rods are provided in the reactor in addition to the twelve operational control rods. The safety rods are not a part of the normal operational control system for the reactor. The safety rods are always in the reactor (in their most reactive position), and they are designed to function when the control rods are disconnected from their drives. The primary purpose of the safety rods is to provide "available negative reactivity" when the reactor is shut down and the control rods are disconnected. They provide a safety factor during reactor loading operations. The safety rod drive mechanisms are completely independent of the fuel handling systems. The reactor cannot be operated nor can the fuel loading equipment be operated unless the safety rods are in the up position. They are actuated by low level detectors separate from the normal operational control system.

The two safety rods are connected beneath the reactor to a horizontal bar which is connected to two vertical shafts extending upward through the biological shield. Each shaft is coupled by means of a rack and pinion drive to a single drive system. Appropriate shafts and gearing couple these systems together so that they operate as a unit. The rods are raised at a slow speed and drop under the force gravity. Actuation is accomplished by release of a clutch, and deceleration is accomplished by means of hydraulic snubbers.

D. Fuel Handling System

"Fuel handling" includes: removing the subassembly from the reactor, transferring it to the storage rack, and after a 15-day cooling period (for fission product decay), removing it from the primary tank. The fuel-handling system (Fig. 16) consists of the reactor gripper mechanism, the hold-down mechanism, the transfer arm, and the storage rack. The reactor gripper mechanism and the hold-down mechanism are located in the small rotating plug which is, in turn, eccentrically located in the large rotating plug.

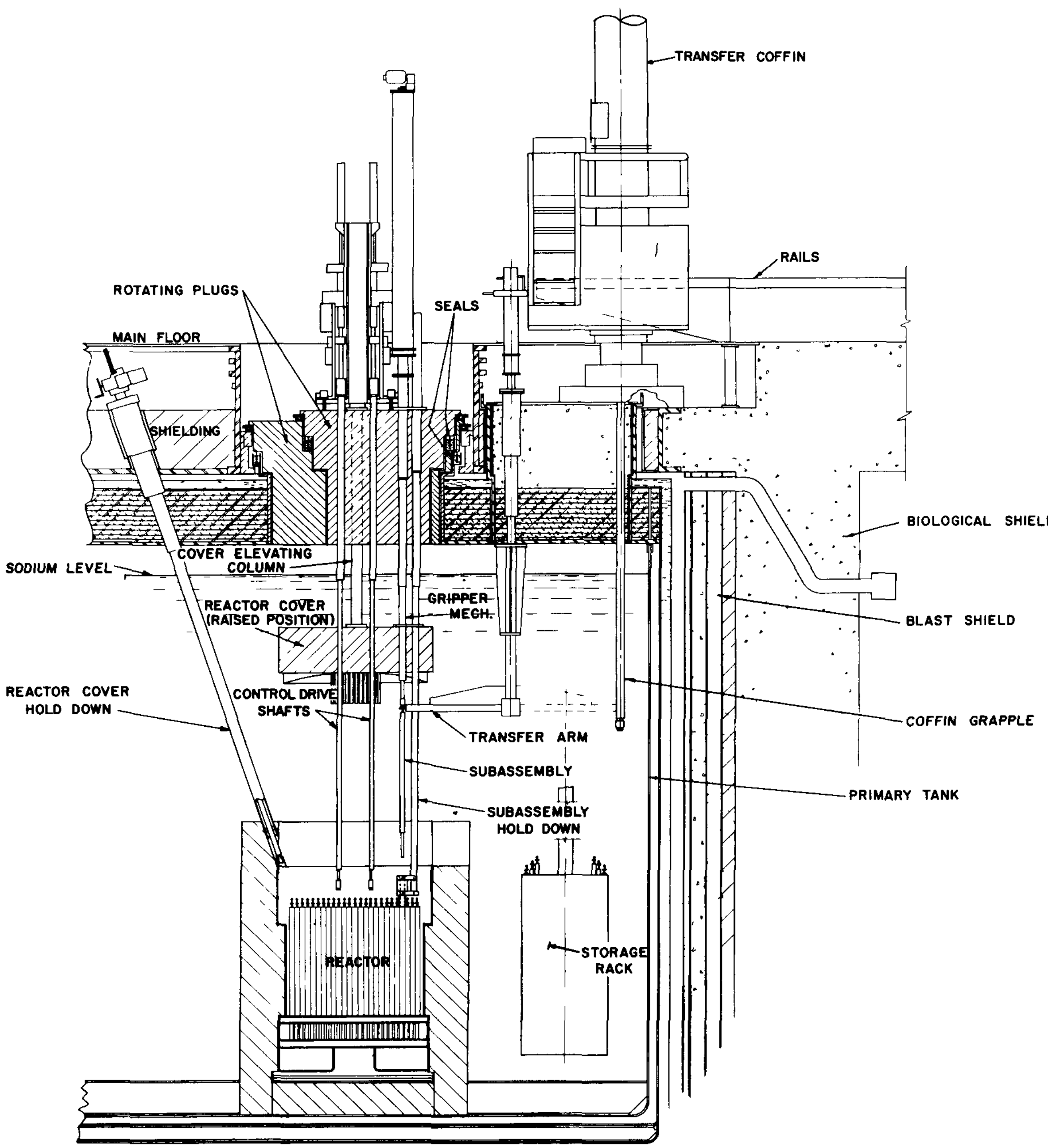


Fig. 16. EBR-II Fuel Handling System

Rotation of the two plugs is employed to position the gripper over the desired location in the reactor, and to position the gripper at the "transfer position." The reactor cover is also supported by the small plug and rotates with it. The gripper mechanism and hold-down mechanism operate through the cover.

To remove a subassembly, the reactor is properly shut down, the twelve control rods are released from their individual control rod drive mechanisms, the reactor cover hold-down clamps are released, the cover is raised, and the rotating plugs are rotated to the proper location to position the gripper over the desired subassembly. The subassembly hold-down mechanism, consisting of a "funnel-shaped" sleeve, is lowered over the subassembly to be removed, contacting the six adjacent subassemblies, spreading them slightly, and preventing them from moving as the subassembly is removed. The gripper head is lowered, engaging the adapter on the subassembly, and the subassembly is lifted vertically from the reactor.

After lifting the subassembly to the proper elevation, the plugs are rotated to the transfer point. The transfer arm is positioned to receive the subassembly which is then lowered such that the collar of the subassembly adapter fits into a recess on the transfer holding device. The subassembly is now released by the gripper and accepted by the transfer arm.

The transfer arm with the subassembly is rotated through a horizontal arc of about 80° and positions the subassembly above any one of three concentric rows of storage locations in the storage rack. The storage rack is a tank-shaped structure providing 70 storage locations in three concentric rows. It is suspended by a shaft extending through the primary tank cover and can be rotated as well as raised or lowered to different levels in the primary tank. An empty storage location is positioned below the subassembly on the transfer arm, the storage rack is raised, and the subassembly is inserted into the rack. After subassembly transfer, the transfer arm is rotated to a neutral position and the storage rack is lowered.

After cooling for 15 days to permit fission product decay, the subassembly is removed from the storage rack and transferred to the Process Plant. Although the fission product decay heat generation is significantly less than that obtaining at the time of subassembly removal from the reactor, cooling by forced circulation of an inert gas is required upon removing the subassembly from the sodium. Cooling is provided through all subsequent steps involved in the transport and handling of the subassembly until it has been disassembled and the fuel elements separated from the close-packed cluster.

Introduction of a new fuel or blanket subassembly into the reactor is achieved by the same steps in the reverse order.

III. OPERATION AND PERFORMANCE

A. Control of Power System

Control of the power system is centralized in a control room located in the Power Plant Building. Control, in general, is manual. Only the simplest of control functions, or those which might adversely affect facility safety if handled manually, are effected automatically. As examples, control of primary and secondary system coolant flow rates is manual; control of reactor power level is manual during raising or lowering of power, but is automatic at steady state; control of feedwater flow rate and steam pressure is automatic.

In essence, the basic control philosophy for the EBR-II power system consists in providing: (1) control of reactor power level; (2) maintenance of balance between the rates of heat removal effected by each of the major thermal systems, from the cooling tower to the reactor; and, (3) maintenance of essentially complete isolation of the reactor from the effects of turbine-generator load variation.

Control of Reactor Power

Operational control of reactor power is effected by movement of the control rods. Changes in reactor power level are effected by manual adjustment of rod positions. Power level is maintained at steady state by automatic control of the regulating rod; the control being based on sensing of neutron flux level. Upon occurrence of a reactor scram signal, all twelve rods automatically are ejected from the core.

A total of eleven fission counters and ionization chambers are provided which enable sensing of reactor period, as well as neutron flux level, throughout the range from source power to several times full power. From these detectors are derived the signals for measurement and control of reactor power and for initiation of reactor scram in the event of excessively short periods or excessively high power. The detectors are located in eight vertical thimbles at various positions outside the reactor vessel. The thimbles are immersed in the primary tank sodium and are positioned near the outer surface of the radial neutron shield.

Maintenance of Balance Between Major Thermal Systems

The major thermal systems are the primary system, secondary system, and steam system (plus circulating water). Heat energy generated within the reactor is transported by these systems to the turbo-generator wherein it is converted to electrical energy. The heat released in the condenser is absorbed by the circulating water system and dissipated to the atmosphere via the cooling tower. A simplified flow diagram for the major systems is shown in Fig. 2.

Maintenance of balance between the thermal systems consists principally of balancing the heat removal rate of the secondary system with the heat generation rate in the reactor, since the steam system is designed to furnish a constant feedwater temperature to the steam generator at all loads.

even with the turbo-generator inoperative. Any imbalance of these two rates produces a continuous change in the primary tank bulk sodium temperature.

In order to effect proper balance, the following method of control is employed:

1. The primary system flow rate is regulated to provide a predetermined reactor coolant outlet temperature, varying from 900°F at full power to 850°F at very low power. This 50°F variation in reactor coolant outlet temperature is employed in order to maintain a constant steam temperature of 850°F at all power levels.

2. The secondary system flow rate is adjusted such that the primary tank bulk sodium temperature remains constant. Irrespective of reactor power level, the temperature of the cold leg of the secondary system automatically remains between 580 and 610°F, and the temperature of the hot leg of the secondary system remains relatively constant, varying from about 880°F at full power to about 850°F at very low power (reflecting control of the reactor outlet coolant temperature). Thus, the rates of heat removal from both the primary system and the secondary system are approximately proportional to the secondary system flow rate.

3. Feedwater flow rate and temperature to the steam generator are automatically controlled. System pressure is held by a back pressure valve which dumps excess steam directly to the condenser.

Isolation of the Reactor From Effects of Load Variation

An important feature of the EBR-II in regard to reactor stability is the virtual isolation of the reactor from the effects of changes in power system conditions external to the reactor. This is accomplished in the following manner:

1. No automatic control adjustment link exists between reactor power and electrical load demand or any power system operating condition (other than scram).

2. The steam system incorporates a full flow steam by-pass around the turbine to the condenser, a turbine-generator load limiting device, and a regenerative feedwater system which delivers constant temperature feedwater to the steam generator under all load conditions. This arrangement eliminates any effect of change in turbine-generator (electrical) load upon reactor inlet coolant temperature.

3. A primary system is employed with a very large thermal capacity such that the reactor inlet temperature can only change very slowly. For example, with the reactor operating at full power (62.5 mw) and no heat being removed from the primary tank, the rate of bulk sodium temperature rise would be only about 14°F/min.

B. Power Operation

Reactor Heat Generation Distributions

An approximate breakdown of power generation in the various zones of the "clean" reactor is given in the Appendix (Table I). The core generates 53.3 mw or 85.4% of the power.

The radial power density distribution at the center plane of the reactor is shown in Fig. 17.

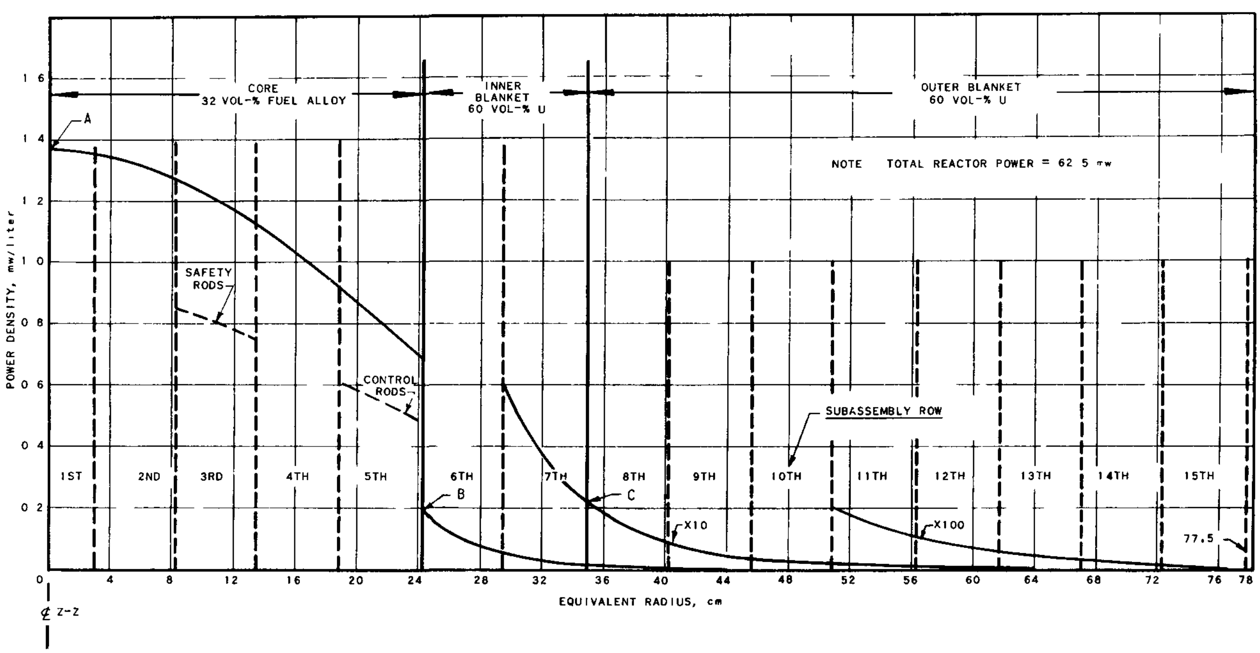


Fig. 17. Radial Power Density Distribution at Center Plane of Clean Reactor (Z = 0).

This distribution is based on 82% insertion of each of the twelve control rods and 100% insertion of the two safety rods. In the control rod and safety rod regions of Fig. 17, two power density distribution curves are presented. The dashed line shows the anticipated power densities in control or safety subassemblies; the solid line indicates the power densities expected in the core subassemblies within these regions.

The radial maximum to average power density ratio of the core at the reactor center plane is 1.33; the effective radial maximum to average ratio over the entire height of the core is 1.32.

The axial power density distribution at the centerline of the core is a typical chopped cosine type within the core region and decreases exponentially in the axial blanket region.

The axial maximum to average power density ratio at the inner edge of the inner blanket is 3.45; at the inner edge of the outer blanket, 2.84.

The axial maximum to average ratio at the centerline of the core for the core section only is 1.17. The effective axial maximum to average ratio over the entire radius of the core for the core section only is 1.16.

Maximum power density in the core is 1.37 mw/liter of core volume. Average power density in the core is 0.893 mw/liter. The ratio of maximum to average power density in the core is 1.53.

These power densities and distributions are based on the clean reactor; however, analysis indicates that only small changes in heat generation distribution are effected by core burnup and plutonium formation, and these changes have been allowed for in calculation of maximum fuel and blanket temperatures. They also include the contributions of local absorption of gamma rays.

Reactor Temperature Distribution

The maximum mean reactor outlet coolant temperature is held at a high level by orificing all subassembly rows except rows 1 and 2 (numbered radially outward from the center of the core). The degree of orificing employed in each row is based on one or more of these limitations (depending upon the row): maximum permissible fuel alloy or blanket uranium temperature; maximum permissible coolant temperature at subassembly outlet; minimum acceptable orifice size.

In addition to flow distribution, another consideration affecting calculated temperature distributions within the reactor is the degree of uncertainty associated with each of the quantities (such as thermal conductivity value, heat transfer coefficient, power density level, etc.) entering into the temperature calculations. Each quantity is analyzed separately, a degree of uncertainty estimated, a factor assigned, and the effect of each factor upon the various temperature differences estimated. The uncertainty factors employed are:

Coolant Temperature Rise	1.18
Coolant Film Temperature Difference	1.36
Clad Temperature Difference	1.31
Bond Temperature Difference	1.25
Fuel Temperature Difference	1.32

In many of the figures referred to in this section, two temperature curves for the same conditions are shown: one which includes the use of uncertainty factors in the calculations, and one which does not. The temperature distributions which involve the use of these factors are considered very conservative. The temperature distributions based on nominal calculations (without uncertainty factors) are those more representative of the average conditions expected to exist within the reactor.

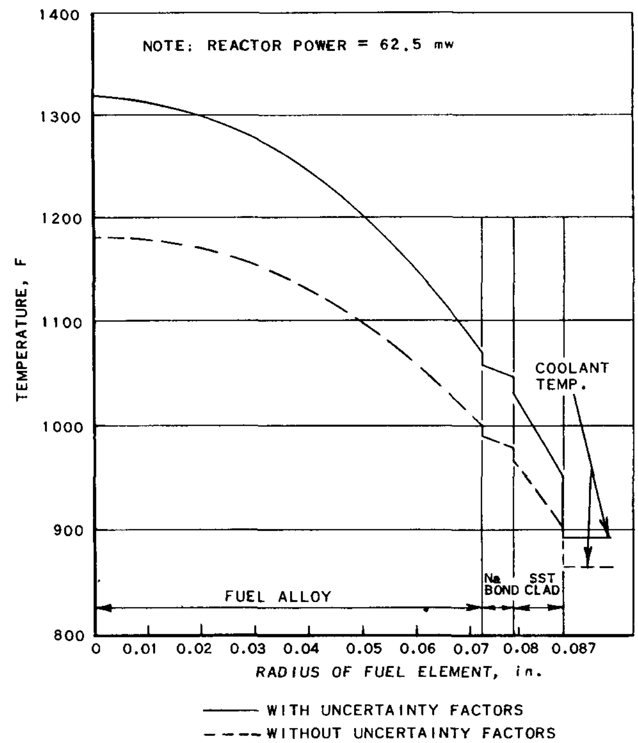


Fig. 18. Radial Temperature Distribution Through a Fuel Element at Point of Maximum Fuel Temperature.

Based on the above considerations, each type of fuel and blanket element is designed to cool under the most severe conditions to which that type of element is exposed. The most severe conditions within each major reactor zone occur in the first, sixth, and eighth subassembly row (centerline of core, inner edge of the inner blanket, and inner edge of the outer blanket, respectively).

Radial temperature distributions through a fuel element at the point of maximum fuel alloy temperature in the reactor are shown in Fig. 18. The effect of uncertainty factors on temperature distribution is apparent.

The mixed mean coolant temperature rise through each subassembly is shown in Fig. 19.

Power Cycle Operating Conditions

The contemplated steady-state operating temperatures and coolant flow rates in the principal heat transfer systems at full power are shown in Fig. 2 and described below.

1. Primary System

The primary sodium coolant flow rate through each of the main pumps is approximately 4250 gpm at a head of 60 psi; total flow is about 8500 gpm. Flow through the reactor totals 8200 gpm; the remaining 300 gpm representing leakage back to the primary tank bulk sodium through the ball-seat disconnects and the subassembly hold-down devices. Of this total flow

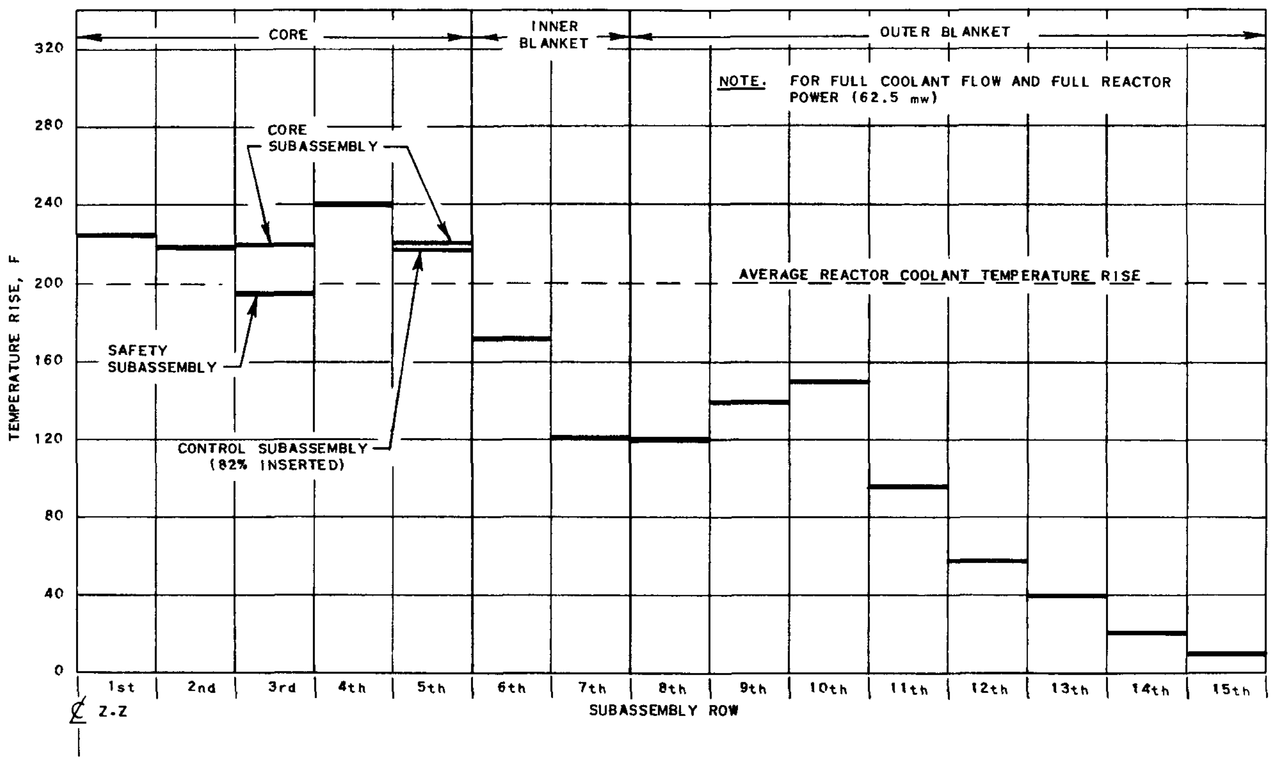


Fig. 19. Average Coolant Temperature Rise Through Each Subassembly

through the reactor, 7000 gpm flows from the high-pressure plenum through the core and inner blanket subassemblies, 700 gpm flows from the low-pressure plenum through the outer blanket subassemblies, and 500 gpm flows through the clearance spaces between subassemblies.

The mixed mean primary coolant temperature in the reactor outlet plenum chamber is 900°F. The coolant enters the heat exchanger at approximately 900°F, since the heat loss between the reactor and heat exchanger is very small. After passing through the heat exchanger, the coolant returns to the bulk sodium with a temperature slightly different from that of the bulk sodium. This small temperature difference is required to maintain the bulk sodium temperature constant at 700°F, because of extraneous heat losses and heat gains by the bulk sodium; viz., gains due to gamma heating of the bulk sodium and coolant leakage past the control drives in the reactor tank cover, and losses due to the shutdown coolers and to the shield cooling air.

2. Secondary System

The secondary sodium system includes a sodium to sodium heat exchanger located in the primary tank, a sodium to water-steam heat exchanger (steam generator) located in the Boiler Plant, an A-C electromagnetic linear induction pump and a surge tank located in the Sodium Plant, and interconnecting piping.

At full power, a flow rate of 6050 gpm at a head of about 60 psi is provided by the pump. The sodium coolant enters the heat exchanger at 610°F and leaves at 880°F. Since the heat loss in the interconnecting piping is small, the sodium enters the superheater section of the steam generator at approximately the same temperature, 880°F. The coolant leaves the steam generator at 610°F.

The steam generator is a natural recirculating type consisting of evaporator and superheater section with a single steam drum. Each section is composed of a number of vertical units. Each evaporator unit is connected to the steam drum by a single downcomer and riser. Dry and saturated steam from the steam drum passes downward through the superheater units.

Both evaporator and superheater units use single length, double-walled tubes. No third fluid annulus or monitoring system is employed. The outer tube is welded to the shell tube sheet and the inner tube to an external tube sheet. These tube sheets are positioned a few inches apart and connected only by the tubes. There are no welds in these units which separate sodium from water or steam.

3. Steam System

Feedwater at 550°F is supplied to the steam drum of the steam generator at a rate of 249,000 lb/hr during full power operation. Saturated steam at 580°F is generated within the evaporator. This steam passes through conventional moisture separators in the steam drum and then is raised to a temperature of 850°F in the superheater. Steam conditions at the turbine throttle are 1250 psig, 850°F with a flow rate of about 199,000 lb/hr. The remaining steam is employed for direct feedwater heating, for driving the feedwater pump turbine, and for maintenance of by-pass flow around the turbine to the condenser. The turbine exhausts 145,500 lb/hr of steam (moisture content 14%) to the condenser operating at $1\frac{1}{2}$ in. Hg pressure. The condenser, a deaerating type, employs a circulating flow rate of 23,600 gpm at 70°F inlet and 82°F outlet temperatures respectively.

The feedwater heating system consists of a blowdown cooler, two feedwater heaters employing steam extracted from the turbine, one deaerating heater which utilizes steam from the exhaust of the feedwater pump turbine, and a high pressure heater using steam supplied directly from the main steam line (1250 psig, 850°F). This system is designed to supply feedwater to the steam generator at a constant temperature regardless of load conditions or mode of plant operation.

The steam system flow diagram, shown in Fig. 2, includes 5000 lb/hr of steam by-passed to the condenser at full load (62.5 mw of heat). Under these conditions of operation, the generator produces 20,700 kw gross output.

IV. RELIABILITY AND SAFETY

Much of the EBR-II design effort has been directed toward attainment of maximum reliability and safety. A large number of features are incorporated which promote reactor operational reliability, minimize probability of occurrence of a destructive nuclear accident, and assure containment of radioactivity which would be released in the very remote event that such an accident should occur. Only a few of the pertinent major provisions are reviewed here. A more complete treatment is presented in the EBR-II "Hazards Summary Report."(3)

A. Electrical System Provisions

The electrical system incorporates two independent transformer ties to a closed power loop (138 kv) served by two separate utility companies. Both transformers are connected to a sectionalized main bus (13.8 kv), the tie breakers being closed during normal operation. Also connected to one of the two main bus sections is the EBR-II generator. Thus, the plant is equipped with three sources of main (13.8 kv) electrical power. To further the main power system reliability, duplicate circuits to each of the lower voltage systems (2400 and 480 v) are provided from auxiliary transformers fed by both of the main bus sections. In addition, a selective, protective relaying scheme is used to effect automatic isolation of a faulted portion of the system with no interruption of power.

An automatically starting, 400 kw diesel-generator set is incorporated to supply power for operation of necessary or desirable auxiliaries in the remote event of a main power outage. This source is backed up by an additional 75 kw diesel-generator set connected only to the two critical auxiliaries: the nuclear instrument thimble cooling system, and the biological shield cooling system. This latter set also starts automatically upon outage of main power, but comes on the line only if the large set fails to start within a pre-determined period of time.

The system described above, with three sources of main power, duplicate circuits, protective relaying, and two emergency generators, is thought to assure an exceptionally high degree of continuity and reliability of electrical power supply.

B. Coolant Availability Provisions

The unique arrangement of the EBR-II primary system represents a most important safety feature in that it virtually guarantees the availability of coolant to the reactor under all circumstances. The complete system is contained in the primary tank, as shown in Fig. 20. All of the primary system components, including the reactor, coolant pumps, primary piping,

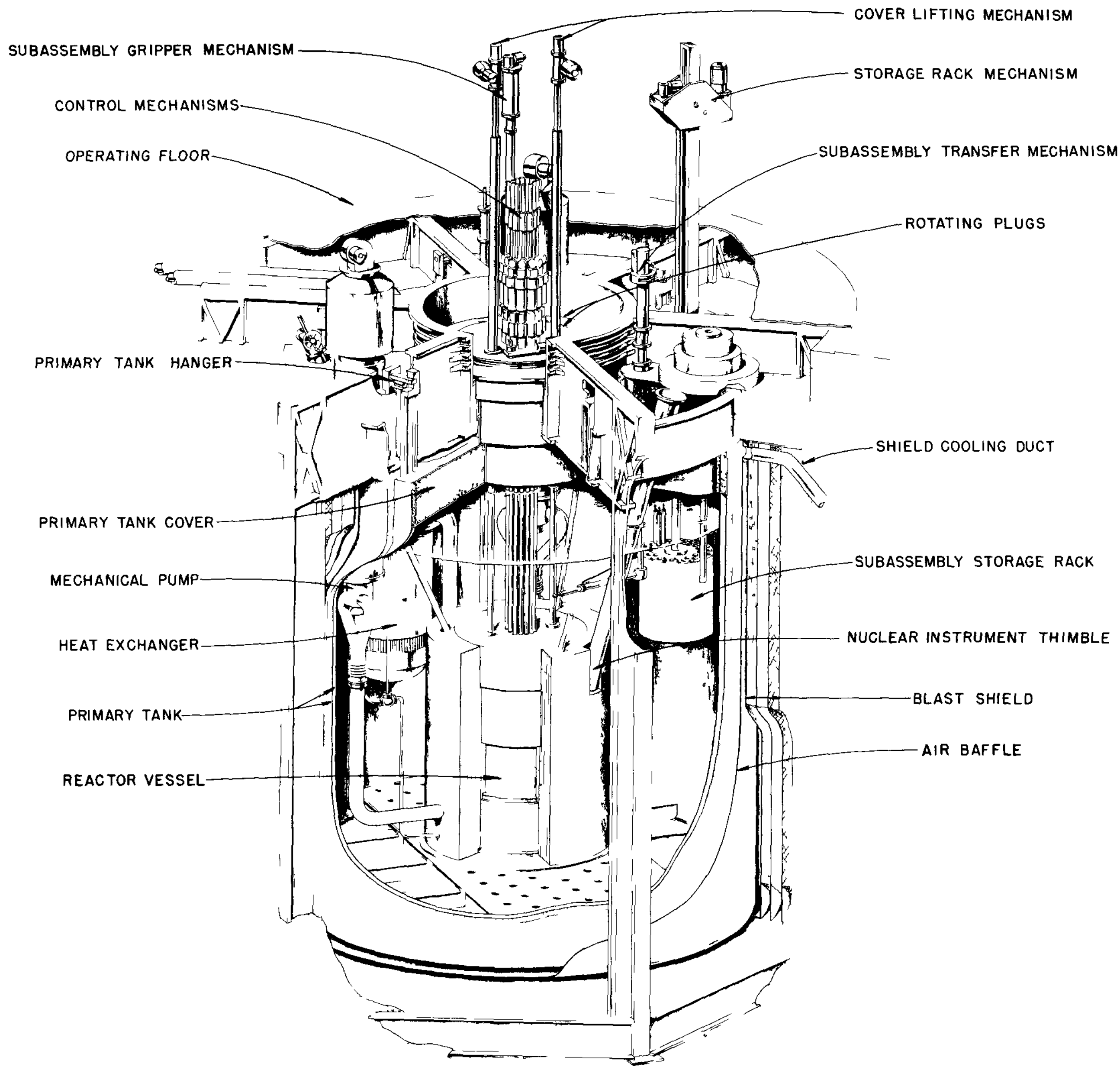


Fig. 20. EBR-II Primary System

heat exchanger, and fuel transfer and storage system are submerged in the large bulk volume (80,000 gal) of sodium within this tank. All penetrations are confined to the tank cover; none exist in the walls or bottom. The few auxiliary systems which communicate with the primary coolant are designed to preclude any possibility of syphoning. The tank itself is double-walled; in respect to confinement of primary system coolant, it may be considered to consist of two concentric, independent tanks. Further, the cavity in which the primary tank is positioned is fitted with a steel liner which constitutes, in effect, a third tank. The free volumes between the various tanks are such that even if the first two tanks were to leak uncontrollably, the free surface of the coolant then being confined by the third tank would lie several feet above the top of the reactor. In view of these provisions, loss of reactor coolant appears virtually impossible.

C. Shutdown Cooling Provisions

The means provided for dissipation of fission product decay heat subsequent to reactor shutdown are believed to be exceptionally reliable. Two steps are involved: providing sufficient coolant flow through the reactor properly to remove the decay heat; and, eventual transfer of this heat from the coolant.

Providing Coolant Flow

After reactor shut down (including scram) the two primary pumps will normally continue to operate and flow will be reduced gradually before stopping the pumps. If one pump stops the other pump supplies far more coolant than required.

If both pumps are inoperative, the auxiliary pump provides more than adequate flow under any shut down condition. In the event of a total power failure, the auxiliary pump is driven by the storage battery power supply, in which case the flow gradually decreases as the batteries discharge.

The third and basic source of flow is natural convetion. As indicated in Fig. 2., the relative elevations of the primary system components promote a large thermal driving head. Irrespective of heat exchanger temperature distributions (or secondary system flow rate), the head developed is sufficient to produce adequate coolant flow. Accordingly, even if all electric power including the auxiliary pump battery supply were lost, or if all three primary system pumps failed simultaneously, the reactor would still continue to be cooled properly.

Transfer of Heat from the Primary Coolant

Heat is removed from the coolant emerging from the reactor by one or both of these means: (1) by transfer to the secondary system; (2) by transfer (through mixing) to the bulk coolant within the primary tank.

All heat transferred within the heat exchanger to the secondary system is dissipated to the atmosphere via the steam system and cooling tower. The exact amount of heat so rejected is dependent upon the secondary system flow rate. The natural convection flow normally existing in the secondary system is sufficient to remove the fission product decay heat. An amount considerably in éxcess of the maximum reactor decay heat can be removed by operation of the secondary system pump. This would result, of course, in continuous lowering of the bulk coolant temperature.

All decay heat not transferred to the secondary system appears within the bulk coolant of the primary tank. A portion of this heat is dissipated as parasitic losses to the room, to the instrument thimble cooling system, and to the biological shield cooling system. The remainder is dissipated to the atmosphere by two "shutdown coolers" operating in conjunction with two air-fin heat exchangers.

The shutdown coolers (Fig. 21) are immersion-type, bayonet heat exchangers of 250 kw capacity. Each consists of two concentric pipes approximately 26 ft long, the outer pipe being closed at its bottom end. The cooler is positioned within a vertical thimble immersed in the bulk sodium of the primary tank. A thermal bond of sodium is provided in the space between the cooler and the thimble. NaK enters the inner pipe of the cooler at the top and flows downward to the bottom end of this pipe where it reverses direction and enters the annulus. The flow is then upward through the annulus, where heat transfer from the bulk sodium to the NaK occurs. Leaving the cooler, flow is upward to a finned-tube, NaK-to-air heat exchanger located in a dampered air stack outside the Reactor Building. The NaK is cooled within this exchanger and then returns to the inlet of the cooler.

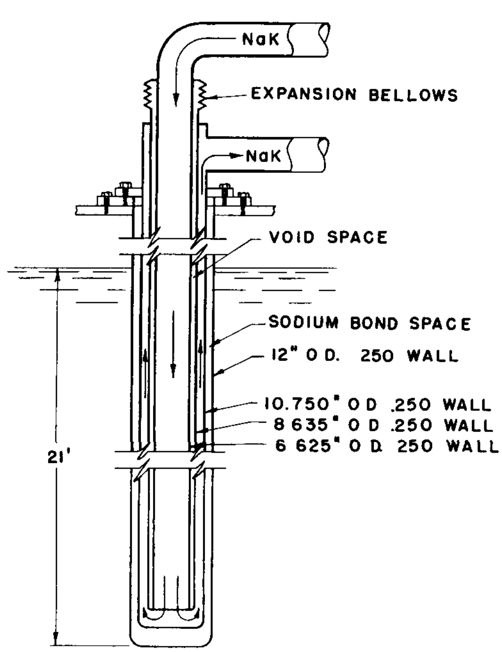


Fig. 21. Shut Down Cooler

The rate of heat release from the system is controlled by the position of the stack damper. Normally, the damper is actuated by automatic control. During reactor operation, the damper is closed and a minimal flow of NaK obtains. When the damper opens subsequent to reactor shutdown, the thermal heads on both the NaK and air sides increase. This gives rise to increased flow rate of both fluids which, in turn, results in increased rate of heat removal from the bulk sodium. This method of operation prevents the freezing of NaK in cold weather, provides for positive starting when the damper is opened, and minimizes possible thermal shock in the system.

The system is designed for maximum reliability and simplicity. The salient feature of the system is the complete independence of any external power source. All fluid flow is by natural convection.

Because of the very large total thermal capacity of the bulk sodium and submerged components, any failure or maloperation of the shutdown coolers produces only an extremely small rate of change of primary bulk sodium temperature. Consequently, ample time is available for corrective action, even under the worst conditions. In Fig. 22, bulk sodium temperature is shown as a function of time after reactor shutdown from full power with two, one, or neither of the shutdown coolers in operation.

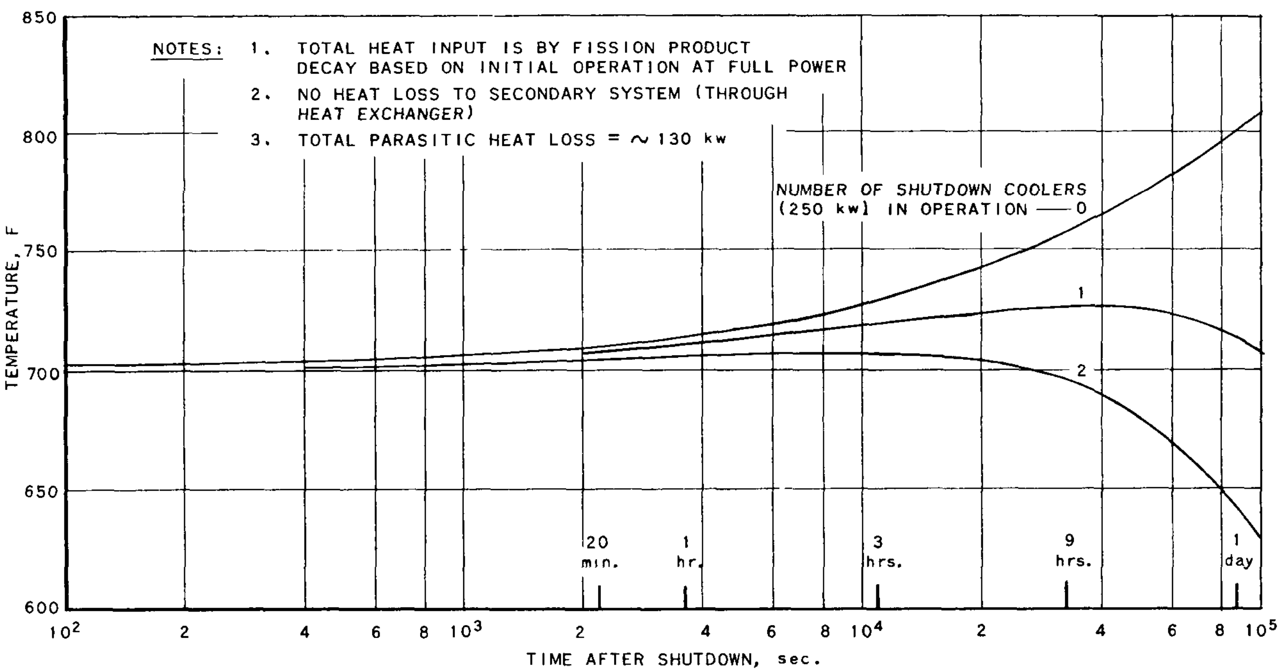


Fig. 22. Primary Tank Bulk Sodium Temperature vs. Time After Shutdown

D. Containment Provisions

Containment of the EBR-II is provided to preclude release of fission products and/or plutonium from the Reactor Plant in the unlikely event of a major nuclear accident. Two echelons of containment are incorporated. The first, termed the "primary containment system," surrounds the primary tank in which the reactor is submerged. Its function is to contain the effects of a nuclear energy release without breaching. The second, termed the "building containment system," surrounds both the primary containment system and the remainder of the Reactor Plant. The function of this system is to localize within the reactor building the effects of a possible sodium-air reaction energy release. A detailed description of the containment provisions is given in another paper prepared for this conference.[4]

Primary Containment System

Design of the primary containment system is based upon an assumed magnitude of nuclear energy release equivalent to the detonation of 300 lb of TNT. The system consists essentially of a cylindrically shaped "pressure vessel" which surrounds the primary tank. Over-all configuration of the vessel may be seen in Fig. 20. The wall of the vessel is formed by the specially reinforced radial biological shield. The top closure is formed by the top structure of the primary tank support structure, together with the additional structure required for support of the primary system component plugs and the top biological shielding. The bottom closure is formed by a reinforced concrete structure which employs for its main beams the bottom structure of the primary tank support structure. Top and bottom closures are tied together by six peripherally positioned columns. These columns as well as the remainder of the primary tank support structure, are of T-1 steel (yield strength, 90,000 psi; ultimate tensile strength, 105,000 psi).

Wall material is ordinary density concrete. Nominal wall thickness is 6 ft, with the columns of the support structure positioned within the innermost 3 ft. The major concrete reinforcement of intermediate grade billet-steel bars is located within the outermost 3 ft. The only penetrations through the wall are those of the shield cooling air ducts, a series of pipes (8 in. dia) extending approximately radially through the wall in a horizontal plane immediately below the bottom surface of the vessel top closure.

The wall is lined on its inner surface with a "blast shield" of 2 ft thickness. The purposes of this shield are to protect the wall from shock wave and to enable absorption of an appreciably large fraction of the total nuclear energy release. A laminated structure employing 3 layers of absorption material is used. The first layer is vermiculite concrete of 23 lb/ft^3, the second is aerated concrete of 16 lb/ft^3, and the third is Celotex of standard specific weight. The various layers are separated by continuous steel cylinders of approximately 3/8 in. thickness.

The top closure main structure consists of 6 radial beams of $6\frac{1}{2}$ ft in depth tying into a central ring (12 ft dia) for accommodation of the large rotating plug, and a reinforcement ring ($27\frac{1}{2}$ ft dia). This structure is criss-crossed with numerous additional beams which provide support for the many primary system components hung from, or penetrating through, the structure. A continuous steel plate of 1 in. thickness is incorporated across the bottom of the structure. The lower 3 ft of the closure is filled with heavy concrete to form part of the top biological shield. The remaining depth is utilized for passage of pipes and lines, and for additional shielding.

Effective missile protection is provided. More than adequate protection from missiles originating within the primary tank bulk sodium, is afforded by the wall and top and bottom closures of the pressure vessel. Only missiles which conceivably could originate within the top closure, such as control rod drive shafts, require consideration. Protection against the latter is provided by a missile shield of reinforced concrete which lines the entire inner surface of the reactor building shell.

Two features inherent in the design of the primary containment system should be specially noted, in that they represent two of the primary factors responsible for the system effectiveness:

1. The presence of a very large volume of sodium immediately surrounding the reactor. Because of the rapid absorption of pressure wave energy within this bulk sodium by the waste heat process, the fraction of total energy released available for loading or destruction of the containment system is greatly minimized.

2. The presence of the free surface of the primary tank bulk sodium. Because of this free surface, the maximum magnitude of the pressure developed and exerted on the bottom of the rotating plug, as well as on the bottom of the primary tank cover (or the pressure vessel top closure), is greatly minimized. The main components of this pressure are the primary tank blanket argon gas shock and the dynamic pressure loading effected by sodium spray. Based on a 300 lb TNT equivalent energy release, the former may be shown to amount to less than 2 atmospheres (abs), to last for not more than 0.002 sec, and not to coincide time-wise with the sodium spray pressure. The latter pressure may be shown to be of magnitude less than 4 atmospheres (abs) and of duration much less than 0.002 sec.

The primary containment system as described above is capable of withstanding, without breaching, a nuclear energy release within the reactor core equivalent to the detonation of 300 lb of TNT. Although production of missiles of significant energy is improbable, the reinforced concrete missile shielding within the building readily stops the most highly energetic missile possible. However, small amounts of sodium vapor would probably escape from the "pressure vessel" into the building atmosphere through the shield air cooling ducts, from ruptured sodium pipes within the top closure, etc.

Building Containment System

The building containment system consists of a carbon steel "building shell" enclosing the primary containment system and the remainder of the Reactor Plant. Figure 23 gives an over-all view of the shell design. The shell is cylindrical, with a hemispherical top closure and a semi-ellipsoidal bottom closure. Inside diameter is 80 ft; total height is approximately 140 ft, about 48 ft of which is below grade. Thickness of the cylindrical section is 1 in., and thickness of the closures is commensurate with this in respect to strength. Design of the shell is based on maximum internal pressure of 24 psig. The shell is to be pneumatically pressure tested at 30 psig.

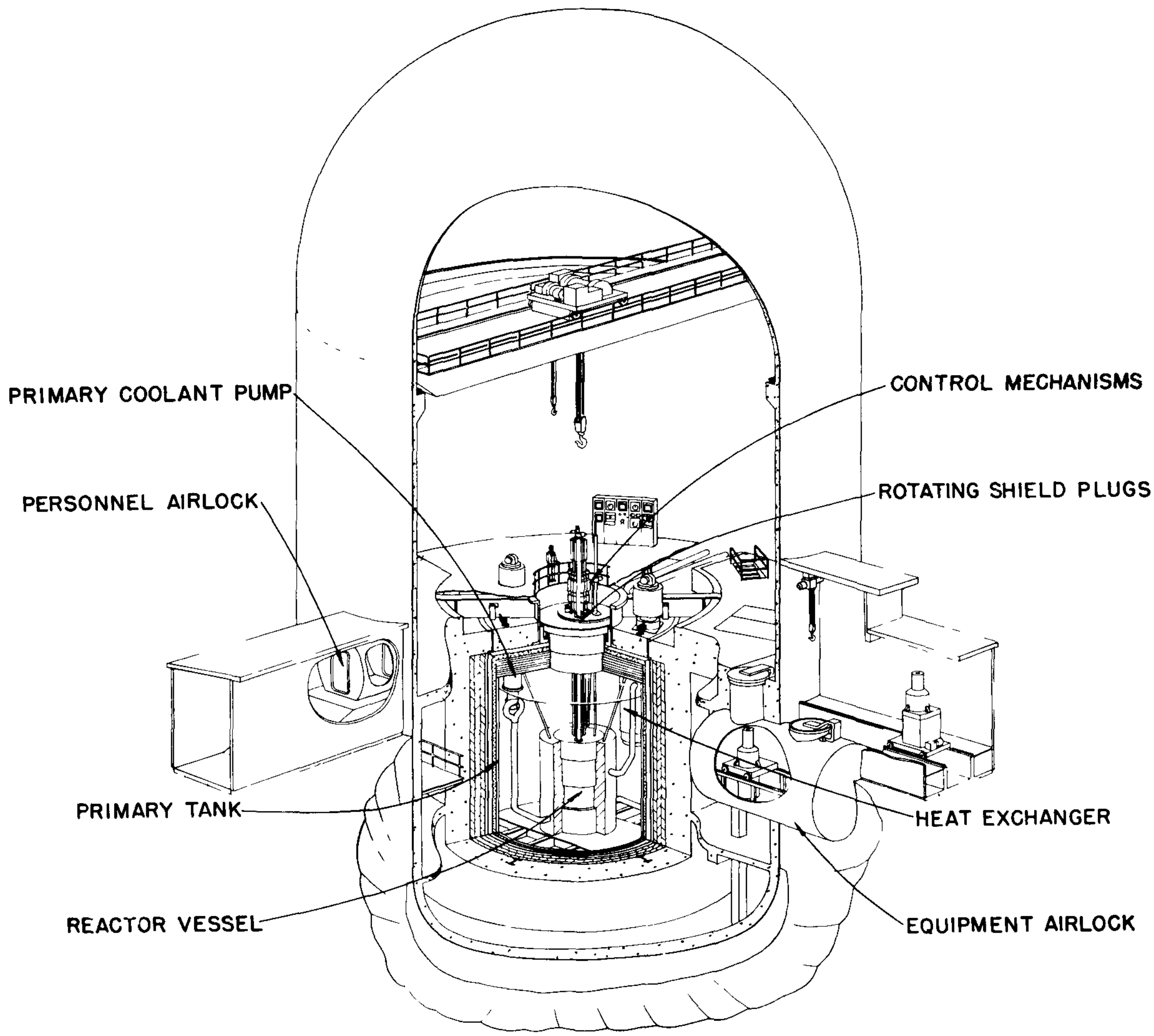


Fig. 23. EBR-II Reactor Plant

A large number of openings through the shell are required for personnel, equipment, ventilating air, sodium pipe, electrical conductor, and other access. All openings employ gas tight seals, either of the metal-to-metal type or of an organic type suitably protected from the high temperatures of building gas which possibly could be realized in the event of a serious accident. All openings are designed so as not to detract from the strength of the building shell and so as to be capable of sustaining the same building pressure as the maximum containable by the shell itself. At final acceptance, total building leakage rate with internal pressure of 24 psig will not be more than 1000 cu ft per day.

As discussed earlier, the design objective of the primary containment system is to withstand a nuclear accident without breaching. There of course exists some assumed magnitude of nuclear energy release sufficiently large to effect breaching of the primary containment system. With this assumption, appreciable amounts of sodium could be ejected into the building atmosphere. From information presented in another paper prepared for this conference,[5] it can be estimated that ejection of some 3,000 lbs of sodium with highly efficient dispersion, or considerably larger amounts with a more realistic degree of dispersion, would be required to produce a building pressure in the vicinity of 24 psig (and gas temperature of about 1200°F). The building containment system, designed for maximum stress of 15,000 psi at 24 psig internal pressure, easily contains such pressure (gas temperature being no problem, since the allowable stress remains constant up to 650°F and the maximum shell temperature realized would obviously be considerably lower than this value). After sufficient reduction in gas temperature occurs, the building pressure tends to fall below atmospheric; however, an automatic nitrogen bleed-in system maintains the maximum pressure differential at a safe value.

V. CONCLUSION

The EBR-II is an experimental plant, however, it has been designed as a prototype central station type power plant insofar as possible. It will not, of course, produce economically competitive power, however, it will establish the engineering feasibility of many important plant parameters. It is expected that the cost information and performance data developed will permit extrapolation to full size plants, and thereby establish economic feasibility as well.

In addition to the reactor and power cycle, the EBR-II will establish the technical feasibility of a fuel cycle which appears to be particularly attractive for fast reactors. Although this cycle is adaptable to U^{235} - U^{238} and Pu^{239} - U^{238} fuel alloys the use of plutonium is of paramount interest. Because fast reactors utilize plutonium much more efficiently than thermal reactors, they will become increasingly important in a nuclear power economy as thermal power reactors produce increasing quantities of plutonium. This will only be true, however, if plutonium fueled fast power reactors are developed which meet the peculiar technological and economic requirements presented by this system. The EBR-II is directed toward this objective.

APPENDIX

Table I

SUMMARY: EBR-II DESIGN DATA

General		
Heat Output	62.5	mw
Gross Electrical Output	20	mw
Primary Sodium Temperure, to reactor	700	F
Primary Sodium Temperature, from reactor	900	F
Primary Sodium Flow Rate, through reactor	8200	gpm
Primary Sodium Maximum Velocity, in core	26	fps
Primary System Sodium Capacity	86,000	gal
Secondary Sodium Temperature, to heat exchanger	610	F
Secondary Sodium Temperature, from heat exchanger	880	F
Secondary Sodium Flow Rate	6050	gpm
Steam Generator		
Output	249,000	lb/hr
Steam Temperature	850	F
Steam Pressure	1310	psig
Feed-Water Temperature	550	F
Turbine Throttle Conditions		
Steam Flow	199,000	lb/hr
Steam Temperature	850	F
Steam Pressure	1250	psig
Reactor Data		
Core and Blanket Dimensions		
Core Equivalent Diameter	19.04	in.
Inner Blanket Equivalent O.D.	27.46	in.
Outer Blanket Equivalent O.D.	61.5	in.
Core Composition		
Fuel Alloy	31.8	vol-%
Stainless Steel (Type 304)	19.5	vol-%
Sodium	48.7	vol-%
Control and Safety Rod Composition (Fuel Section)		
Fuel Alloy	21.3	vol-%
Stainless Steel (Type 304)	20.8	vol-%
Sodium	57.9	vol-%

Table I (Contd.)

Reactor Data (Contd.)

Upper and Lower Blanket Composition		
Uranium (depleted)	32	vol-%
Stainless Steel (Type 304)	20.4	vol-%
Sodium	47.6	vol-%
Inner and Outer Blanket Composition		
Uranium (depleted)	60	vol-%
Stainless Steel (Type 304)	17.6	vol-%
Sodium	22.4	vol-%
Subassemblies		
Core	47	
Control (Rod and Thimble)	12	
Safety (Rod and Thimble)	2	
Inner Blanket	66	
Outer Blanket	510	
Total	637	
Configuration	hexagonal	
Dimension across flats	2.290	in.
Hexagonal Tube Thickness	0.040	in.
Structural Material	304 SS	
Lattice Spacing (Pitch)	2.320	in.
Clearance between subassemblies	0.030	in.
Fuel Elements (Pin-Type, Sodium Bonded)		
Fuel Pin Diameter	0.144	in.
Fuel Pin Length	14.22	in.
Fuel Tube O.D.	0.174	in.
Fuel Tube Wall Thickness	0.009	in.
Thickness Na Bond Annulus	0.006	in.
Elements per subassembly	91	
Upper and Lower Blanket Elements (Pin-Type, Sodium Bonded)		
Blanket Pin Diameter	0.316	in.
Blanket Pin Length (Total)	18	in.
Blanket Tube O.D.	0.376	in.
Blanket Tube Wall Thickness	0.022	in.
Thickness Na Bond Annulus	0.008	in.
Blanket Elements per subassembly (each end)	19	
Control and Safety Rods		
Configuration	hexagonal	
Dimension across flats	1.908	in.

Table I (Contd.)

Reactor Data (Contd.)

Control and Safety Rods (Contd.)		
Fuel Elements	same as core subassembly	
Fuel Elements per rod	61	
Inner and Outer Blanket Elements (Pin-Type, Sodium Bonded)		
Blanket Pin Diameter	0.433	in.
Blanket Pin Length (Total)	55	in.
Blanket Tube O.D.	0.493	in.
Blanket Tube Wall Thickness	0.018	in.
Thickness Na Bond Annulus	0.012	in.
Blanket Elements per Subassembly	19	
Fuel Alloy (Enriched U-Fissium)		
Total Core Loading	363	kg
U^{235} Enrichment	49	%
Critical Mass - U^{235}	170	kg
Fuel Alloy Composition: (Fissium)		
Uranium	95.0	wt. %
Zirconium	0.2	wt. %
Molybdenum	2.5	wt. %
Ruthenium	1.5	wt. %
Rhodium	0.3	wt. %
Palladium	0.5	wt. %
Fertile Blanket Material (Depleted Uranium)		
Total Blanket Loading	28,100	kg

Nuclear Data

Total Fissions per cc/sec, at Center of Core	4.4×10^{13}	
Neutron Energy Distribution at Center of Core		
Flux above 1.35 Mev	0.8×10^{15}	n/(cm^2)(sec)
Flux below 1.35 Mev	2.9×10^{15}	n/(cm^2)(sec)
Total Neutron Flux	3.7×10^{15}	n/(cm^2)(sec)
Prompt Neutron Life Time	8×10^{-8}	sec

Reactor Control

Power Coefficient	-3.2×10^{-5}	$(\Delta k/k)$/mw
Doppler Effect - Average	$+0.04 \times 10^{-5}$	$(\Delta k/k)$/C
Isothermal Temperature Coefficient	-3.6×10^{-5}	$(\Delta k/k)$/C

Table I (Contd.)

Reactor Control (Contd.)

Total Reactivity (Worth)		
12 Control Rods	0.046	$\Delta k/k$
2 Safety Rods	0.014	$\Delta k/k$
Long-Term Reactivity Effects (From Clean to 2% burnup)		
Burnup of U^{235} in Core	-0.02	$\Delta k/k$
Buildup of Pu in Core	+0.002	$\Delta k/k$
Buildup of Pu in Blanket	+0.007	$\Delta k/k$
Buildup of Fission Products	-0.002	$\Delta k/k$
Irradiation Growth of Fuel (4% Growth)	-0.011	$\Delta k/k$

Heat Transfer

Heat Generation in Reactor		
Core, Control and Safety Subassemblies	53.3	mw
Upper and Lower Blanket	1.2	mw
Inner Blanket	5.2	mw
Outer Blanket	2.6	mw
Neutron Shield	0.2	mw
Heat Generation in Core		
Radial Maximum to Average Power Density at Reactor Center Plane	1.33	ratio
Axial Maximum to Average Power Density at Reactor Center Line	1.17	ratio
Power Density, Average	0.89	mw/liter
Power Density, Maximum	1.37	mw/liter
Power Density, Maximum to Average	1.53	ratio
Specific Power	314	kw/kg
Maximum Heat Flux	1,030,000	$Btu/(hr)(ft^2)$
Average Heat Flux	680,000	$Btu/(hr)(ft^2)$

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* U S GOVERNMENT PRINTING OFFICE 1958 O—468340